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May 18, 2000

5928-00-20187

Mr. John G. Caruso
Division of Reactor Safety
Human Performance and Emergency Preparedness Branch
U. S. Nuclear Regulatory Commission, Region I
475 Allendale Road
King of Prussia, PA 19406-1415

SUBJECT: THREE MILE ISLAND UNIT 1(TMI UNIT 1)
OPERATING LICENSE NO. DPR-50
DOCKET NO. 50-289
SUBMITTAL OF THE REACTOR OPERATOR EXAMINATION AND QUALITY CHECKLIST

Dear Mr. Caruso:

The purpose of this letter is to submit materials required for the Reactor Operator initial licensing examination for your review and approval. The examination materials are being provided to support administration of the Reactor Operator licensing examination at the Three Mile Island Unit 1 facility during the week of June 30, 2000. It is requested that the included materials be withheld from public disclosure until after the examination is complete.

Pursuant to the requirements of ES 201, Section C.1, the written examination and quality checklist were prepared in accordance with the guidelines of ES-401.

The designated point of contact for information regarding the preparation and administration of this examination is Mr.William G. Ogle. He can be contacted at 717-948-2020.

Very truly yours,

John B. Cotton
Vice President, TMI Unit 1

JBC/vlk

cc **w/o enclosures**:
Administrator, Region I - Hubert J. Miller
TMI Senior Resident Inspector - Wayne L. Schmidt
Chief Human Performance and Emergency Preparedness Branch Division of Reactor Safety - Richard J. Conte

File 00003

TAB - EXAM SOURCES

New Exam Questions Report

NRC CRO Licensing Examination
Three Mile Island Nuclear Station
June 2000

Q #	TMI Cognitive Level	Q #	TMI Cognitive Level	Q #	TMI Cognitive Level
002	2	043	2	091	1
003	3	046	3	095	1
004	1	048	1	098	2
005	1	049	2	099	2
006	1	050	3	100	2
007	1	052	2		
009	2	053	2		
010	2	054	3		
012	1	055	3		
013	1	058	1		
014	3	059	1		
015	1	060	1		
017	2	061	2		
018	3	062	3		
019	2	063	1		
020	1	064	1		
021	2	065	2		
022	3	066	1		
023	3	068	2		
025	1	070	3		
026	1	071	1		
028	2	073	1		
031	1	075	3		
032	1	077	1		
033	1	078	2		
034	3	079	1		
035	1	080	1		
036	1	081	2		
037	2	082	1		
038	2	083	1		
039	2	084	2		
040	1	085	3		
041	2	086	2		
042	2	087	2		

Previously Used NRC Exam Questions

NRC CRO Licensing Examination
Three Mile Island Nuclear Station
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Q #	TMI NRC Exam Date
008	1998 SRO EXAM
011	1998 SRO EXAM
047	1998 SRO EXAM
067	1998 SRO EXAM
088	1998 SRO EXAM
097	1998 SRO EXAM
001	FEB 2000 CRO
024	FEB 2000 CRO
027	FEB 2000 CRO
029	FEB 2000 CRO
044	FEB 2000 CRO
045	FEB 2000 CRO
051	FEB 2000 CRO
056	FEB 2000 CRO
057	FEB 2000 CRO
069	FEB 2000 CRO
072	FEB 2000 CRO
074	FEB 2000 CRO
076	FEB 2000 CRO
089	FEB 2000 CRO
090	FEB 2000 CRO
092	FEB 2000 CRO
093	FEB 2000 CRO
094	FEB 2000 CRO
096	FEB 2000 CRO

Nuclear Regulatory Commission
CRO Licensing Examination
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Examination Question Source Report

001	FEB 2000 exam, Q001
002	NEW
003	NEW
004	NEW
005	NEW
006	NEW
007	NEW
008	1998 SRO Exam Question 69
009	NEW
010	NEW
011	1998 SRO EXAM QUESTION 63
012	NEW
013	NEW
014	NEW
015	NEW
016	Modified TMI Bank question.
017	NEW
018	NEW
019	NEW
020	NEW
021	NEW
022	NEW
023	NEW
024	Modified TMI bank question.
025	NEW
026	NEW
027	Bank question. FEB 2000 exam question 60.
028	NEW
029	Bank question. FEB 2000 exam question 19.
030	TMI Exam Bank Question QR5D04-04-Q01
031	NEW
032	NEW
033	NEW
034	NEW
035	NEW
036	NEW
037	NEW
038	NEW
039	NEW
040	NEW
041	NEW
042	NEW
043	NEW
044	Bank question. FEB 2000 question 46.
045	Bank question. FEB 2000 exam question 47.
046	NEW
047	1998 SRO EXAM QUESTION 71
048	NEW
049	NEW
050	NEW
051	FEB 2000 CRO QUESTION 51
052	NEW
053	NEW
054	NEW
055	NEW
056	TMI Exam bank. FEB 2000 CRO Exam Question 76
057	Bank question. FEB 2000 NRC Exam question 32.
058	NEW

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Examination Question Source Report

059	NEW
060	NEW
061	NEW
062	NEW
063	NEW
064	NEW
065	NEW
066	NEW
067	1998 SRO EXAM QUESTION 21
068	NEW
069	FEB 2000 CRO EXAM QUESTION 69
070	NEW
071	NEW
072	FEB 2000 CRO EXAM QUESTION 72
073	NEW
074	TMI Bank. 2000 FEB NRC exam.
075	NEW
076	Bank question. February 2000 RO exam, question 14.
077	NEW
078	NEW
079	NEW
080	NEW
081	NEW
082	NEW
083	NEW
084	NEW
085	NEW
086	NEW
087	NEW
088	1998 SRO EXAM QUESTION 80
089	FEB 2000 CRO EXAM QUESTION 91
090	FEB 2000 CRO EXAM QUESTION 88
091	NEW
092	FEB 2000 CRO EXAM QUESTION 93
093	FEB 2000 CRO EXAM QUESTION 94
094	FEB 2000 CRO EXAM QUESTION 95
095	NEW
096	FEB 2000 CRO EXAM QUESTION 98
097	FEB 2000 CRO EXAM QUESTION 97
098	NEW
099	NEW
100	NEW

TAB - Modified Questions

Modified TMI Exam Bank Questions Report

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016 Plant conditions:
- Reactor Power is 100%.
- The following ICS Stations are in HAND due to a BTU Limit modification in progress:
  FW-P-1A, FW-P-1B
  FW-V-16A, FW-V-16B
  FW-V-17A, FW-V-17B
  MAP alarm H-1-2, OTSG A BTU LIMIT, is actuated
  MAP alarm H-1-3, OTSG B BTU LIMIT, is actuated
- All other ICS stations are in AUTOMATIC

A Loss of ICS HAND Power occurs.

Identify the ONE (1) statement below that describes required operator action(s) for this situation.

A. Stabilize reactor power to match Main Feedwater flow.

B. Trip the Reactor, Turbine and both Main Feedwater Pumps.

C. Place both Feedwater Pumps and all Feedwater Valves into ICS Auto.

D. Transfer both Feedwater Pumps to the Motor Speed Changers in the Control Room.

Correct Answer: B

Parent Question: 15

Parent Question Text

Plant conditions:
- Reactor Power is 42%.
- The following ICS Stations are in HAND due to ICS automatic control problems:
  FW-P-1A, FW-P-1B (FWP flows are matched).
  FW-V-16A, FW-V-16B
  FW-V-17A, FW-V-17B
- All other ICS controls are in AUTOMATIC.

LOSS OF ICS HAND POWER occurs, followed immediately by a TURBINE TRIP.

Identify the ONE (1) statement below that describes procedurally required operator action(s) for this situation.

A. Trip the reactor AND trip both Main Feedwater Pumps.
B. Stabilize reactor power equal to Main Feedwater flow.
C. Transfer control of BOTH Feedwater Pumps to the Motor Speed Changers in the Control Room.
D. Transfer BOTH Main Feedwater Pumps to LOCAL SPEED CONTROL at the feedwater pump control panel.

Question Modification Description

Modified:
- Power level
- Reason for components in HAND
- Replaced a distractor
- Reordered distractors

Added:

BTU Limits alarms
Effect:
The new distractor is plausible, but incorrect due to the BTU Limits

Comments None.

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024 Plant conditions:
- ATWS in progress
- The primary CRO depresses the Manual Reactor Trip pushbutton.

Which ONE (1) of the following sets of conditions would positively ensure that the operator's action was successful AND that ALL CRDMs had power removed?

A. RPS Breaker Trip Lights on ALL 4 RPS Cabinets are bright.

B. RPS Protective Subsystem Lights on ALL 4 RPS Cabinets are bright.

C. All IN-LIMIT lamps on the Diamond Control Panel for Groups 1-7 drop are lit.

D. Alarms G-1-3 (CRD DC Trip Brkr Open) AND G-1-4 (CRD AC Trip Brkr Open) actuate.

Correct Answer: A

Parent Question: 17

Parent Question Text The Control Room Operator MANUALLY trips the reactor. Identify the ONE (1) situation below that confirms ALL the CRD breakers have OPENED.

A. All rods in CRD Groups 1-7 drop, fully inserted into the core.
B. Alarms G-1-3 (CRD DC Trip Brkr Open) AND G-1-4 (CRD AC Trip Brkr Open) actuate.
C. RPS Breaker Trip Lights on ALL 4 RPS Cabinets are energized.
D. RPS Protective Subsystem Lights on ALL 4 RPS Cabinets are energized.

Question Modification Description Stem wording has significant changes. Distracter A of the original question was deleted.

Comments None.

Tab- Answer Distribution

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Answer Distribution Report

A=27 B=24 C=25 D=24

A	B	C	D
003	006	007	001
004	008	011	002
005	013	012	028
009	015	014	034
010	016	017	036
019	018	022	037
020	031	023	039
021	038	025	041
024	040	033	045
026	042	043	048
027	047	044	053
029	050	046	054
030	051	049	056
032	052	062	058
035	055	069	060
063	057	070	066
064	059	076	072
065	061	082	075
067	073	084	077
068	080	086	079
071	083	089	081
074	088	090	092
078	091	093	096
085	097	094	098
087		095	
099			
100			

Tab- Pra /Atog

TMI June 2000 NRC Reactor Operator Exam

The following questions are deemed to have a significant relationship to PRA systems or events.

Question

004

009

011

021

022

023

030

036

054

061

073

074

078

079

084

085

086

087

100

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ATOG Questions

Reference	*Q #*	*TMI Obj.*	*Obj. Description*
ATOG 1210-1, Reactor Trip, Rev. 40 page 4	028	V.E.01.08	Given the plant conditions determine which IF/THEN step(s) of ATP 1210-1 apply and which other procedure(s) will be used in conjunction with the ATP 1210-1.
ATOG 1210-1, Reactor Trip, Rev. 40 page 4	014	V.E.01.08	Given the plant conditions determine which IF/THEN step(s) of ATP 1210-1 apply and which other procedure(s) will be used in conjunction with the ATP 1210-1.
ATOG 1210-1, Rev. 40 page 3.	018	V.E.01.03	Given the inability to accomplish any action/verification, list the action(s) which must be taken. (IF/THEN Statements)
ATOG 1210-10 Abnormal Transient Rules, Guides and Graphs, Rev. 34 page 6.	047	V.E.10.01	Without the use of ATP 1210-10 Abnormal Transient Rules, Guides and Graphs, be able to recite and explain the rules and the basis of the rules contained in the procedure.
ATOG 1210-10, Abnormal Transien Rules, Guides and Graphs, page 15, Rev. 37.	052	V.E.09.06	Given plant conditions and OTSG tube-to-shell Delta T determine what actions, if any, should be taken to limit/control OTSG tube-to-shell Delta T in accordance with ATP 1210-10, Abnormal Transient Rules, Guides, and Graphs.
ATOG 1210-10, Abnormal Transient Rules, Guides and Graphs, page 9 Rev. 37.	049	IV.E.01.06	Describe the effects of Reactor Building temperature and pressure changes on process instrumentation.
ATOG 1210-10, Abnormal Transient Rules, Guides, and Graphs, page 4, Rev. 37.	100	IV.A.09.30	Explain how HPI flow is controlled and under what conditions it may be throttled.
ATOG 1210-10, Abnormal Transients Rules, Guides and Graphs, Rev. 37 page 13	055	0618000501	Following an Emergency Feedwater Actuation, respond IAW ATP 1210-10, Abnormal Transients Rules, Guides and Graphs.
ATOG 1210-10, Abnormal Transients Rules, Guides and Graphs, Rev. 37 page 6	004	V.E.01.02	State and explain the Immediate Action(s) of ATP 1210-1, Reactor Trip, in the required order.
ATOG 1210-6, Small Break LOCA Cooldown, Rev. 28 page 2	022	V.E.08.09	Given plant conditions determine if the RCS is superheated and what actions are required.
ATOG 1210-6, Small Break LOCA Cooldown, Rev.28 page 3	021	V.E.06.10	Given plant conditions determine if entry into ATP-1210-6, Small Break LOCA Cooldown, is required and what actions are to be taken.
ATOG ATP 1210-4, Lack of Primary to Secondary Heat Transfer, Rev. 21, Page 2. Steam Tables.	029	V.E.04.07	Given plant conditions determine if entry is required into ATP 1210-4, Lack of Heat Transfer, and state the required immediate actions.
EP 1202-2, Loss of Station Power, Rev. 49 page 3 ATOG 1210-1, Reactor Trip, Rev. 40 page 3	009	V.D.01.02/V.E.01.08	Given a list of symptoms, identify the event, determine what Immediate Automatic Actions will occur and what Immediate Manual actions are required IAW EP 1202-2. Given the plant conditions determine which IF/THEN step(s) of ATP 1210-1 apply and which other procedure(s) will be used in conjunction with the ATP 1210-1.

Reference	*Q #*	*TMI Obj.*	*Obj. Description*
OS-24, ATOG Theory and Practice, Rev. 1 page 5	027	V.E.05.05	Describe the methods available in ATP 1210-5, OTSG Tube Leakage, for limiting integrated OTSG tube leakage.

Tab- Cognitive Level Justification

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Question Cognitive Level Ratings

Q #	NRC	TMI	Q #	NRC	TMI	Q #	NRC	TMI	Q #	NRC	TMI
001		3	026		1	051		3	076		1
002		2	027		2	052		2	077		1
003		3	028		2	053		2	078		2
004		1	029		3	054		3	079		1
005		1	030		3	055		3	080		1
006		1	031		1	056		1	081		2
007		1	032		1	057		2	082		1
008		1	033		1	058		1	083		1
009		2	034		3	059		1	084		2
010		2	035		1	060		1	085		3
011		1	036		1	061		2	086		2
012		1	037		2	062		3	087		2
013		1	038		2	063		1	088		1
014		3	039		2	064		1	089	C	3
015		1	040		1	065		2	090	M	1
016		3	041		2	066		1	091		1
017		2	042		2	067		1	092	M	1
018		3	043		2	068		2	093	M	1
019		2	044		2	069	C	1	094	M	1
020		1	045		2	070		3	095	M	1
021		2	046		3	071		1	096	C	2
022		3	047		1	072	C	2	097		1
023		3	048		1	073		1	098		2
024		3	049		2	074		2	099		2
025		1	050		3	075		3	100		2

NRC M=5 NRC C=4 **TMI 1=45 TMI 2=34 TMI 3=21**

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TMI Cognitive Level Rating: **Q 001**

Rating Rationale:

Examinee is required to synthesize plant conditions and operator actions to predict dynamic core power distribution effects and to determine acceptability of the actions in accordance with plant emergency operating procedures..

Question:

Sequence of events:
- Plant STARTUP in progress (4 RCPs operating).
- Power escalation stopped at 69% reactor power.
    - Group 7 Rod #1 mechanically stuck, 10 inches below the rest of Group 7.
    - Group 7 Absolute Group Average position indication = 55% withdrawn.
- 90 minutes later:
    - Rod #1 is unstuck, withdrawn and re-aligned with remaining Group 7 rods.
    - Power escalation to 100% is resumed at 30% per hour.

Identify the ONE (1) statement below that describes IF and WHY the above actions are procedurally ACCEPTABLE or UNACCEPTABLE in accordance with EP 1202-8, CRD Equipment Failure.

A. Actions are ACCEPTABLE; adverse power peaking will not occur - the stuck rod was re-aligned with its group within two (2) hours.

B. Actions are ACCEPTABLE; adverse power peaking will not occur - the stuck rod was re-aligned with its group while the group was less than 60% withdrawn.

C. Actions are UNACCEPTABLE; adverse power peaking could occur - the rate of power escalation was too rapid after the stuck rod was re-aligned with its group.

D. Actions are UNACCEPTABLE; adverse power peaking could occur - the stuck rod was (withdrawn) re-aligned with Group 7 with reactor power greater than 60%.

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TMI Cognitive Level Rating: **Q 002**

Rating Rationale:

Requires calculation of core delta-T under abnormal circumstances.

Question:

RC-P-1A has been secured due to a #1 seal failure. It is desired to continue operating at 65% power.

Which ONE of the following is closest to the expected core delta-T for this power level?

A. 22.4

B. 29.9

C. 34.5

D. 39.9

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TMI Cognitive Level Rating: **Q 003**

Rating Rationale:

To arrive at the correct answer the examinee must analyze the plant conditions to determine that a reactor vessel head bubble has formed.

Question:

A Natural Circulation Cooldown was in progress when the following sequence of events occurred:
- MS-V-4A failed open
- cooldown rate increased to 120°F/hour
- Pressurizer level increased from 100" to 250" within a few seconds
- MS-V-4A was taken under manual control
- cooldown rate was reduced to 40°F/hour

Which ONE (1) of the following are the proper actions to take following this sequence of events?

A. Stop the cooldown and maintain RCS temperature constant.
Maintain RCS pressure at the current value or slightly higher.

B. Stop the cooldown and maintain RCS temperature constant.
Continue RCS depressurization within the Natural Circulation Pressure-Temperature Limits curve.

C. Continue cooldown at less than 50°F/hour.
Maintain RCS pressure at the current value or slightly higher.

D. Continue cooldown at less than 50°F/hour.
Continue RCS depressurization within the Natural Circulation Pressure-Temperature Limits curve.

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TMI Cognitive Level Rating: **Q 009**

Rating Rationale:

Requires determination of the event from plant conditions and then determination of manual actions required.

Question:

The following plant conditions exist:
- Substation Panel (SS-1) 230 KV bus meters indicate zero volts
- Main Generator Output Breakers are open

Which ONE (1) of the following is a required operator action?

A. Restore normal makeup and seal injection.

B. Control MS-V-4A/B from the backup loaders.

C. Start the Generator DC emergency seal oil pump.

D. Provide the EF-V-30s a Local Auto setpoint of 50%.

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TMI Cognitive Level Rating: **Q 010**

Rating Rationale:

Question requires evaluation of plant response to events in order to determine the cause of the loss of service water.

Question:

Based on expected plant response, identify the ONE (1) event below that would result in AUTOMATIC isolation of the Nuclear Service Closed Cooling Water to the Reactor Building.

A. Large Break LOCA at the discharge of RC-P-1D.

B. Inadvertent train "B" 4 psig ESAS actuation signal.

C. Loss of Nuclear Service River Water cooling system.

D. Pipe rupture on the NS supply header to the RCP coolers

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TMI Cognitive Level Rating: **Q 014**

Rating Rationale:

Requires analysis of a snapshot of plant data to determine the correct course of action. Correct answer is determined by discrepancy between RCS temperature and main steam pressure.

Question:

A reactor trip has occurred and the crew is verifying the immediate actions of ATP 1210-1, Reactor Trip.

The following plant conditions exist five (5) minutes post-trip:

- RCS Pressure	1900 psig	
	Loop "A"	Loop "B"
- RCS Thot	590°F	591°F
- RCS Tcold	587°F	586°F
- OTSG Pressure	1010 psig	750 psig
- OTSG Level	11"	12"
- RM-A-5	4.79E4 cps	
- RM-A-15	3.62E4 cps	

Which ONE of the following ATOG procedures should the crew GO TO?

A. ATP 1210-2, Loss of 25°F Subcooled Margin

B. ATP 1210-3, Excessive Primary to Secondary Heat Transfer

C. ATP 1210-4, Lack of Primary to Secondary Heat Transfer

D. ATP 1210-5, OTSG Tube Leakage

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TMI Cognitive Level Rating: **Q 016**

Rating Rationale:

Examinee is required to analyze plant conditions and data, and to synthesize the applicable emergency procedure and required actions.

Question:

Plant conditions:
- Reactor Power is 100%.
- The following ICS Stations are in HAND due to a BTU Limit modification in progress:
  - FW-P-1A, FW-P-1B
  - FW-V-16A, FW-V-16B
  - FW-V-17A, FW-V-17B
  - MAP alarm H-1-2, OTSG A BTU LIMIT, is actuated
  - MAP alarm H-1-3, OTSG B BTU LIMIT, is actuated
- All other ICS stations are in AUTOMATIC

A Loss of ICS HAND Power occurs.

Identify the ONE (1) statement below that describes required operator action(s) for this situation.

A. Stabilize reactor power to match Main Feedwater flow.

B. Trip the Reactor, Turbine and both Main Feedwater Pumps.

C. Place both Feedwater Pumps and all Feedwater Valves into ICS Auto.

D. Transfer both Feedwater Pumps to the Motor Speed Changers in the Control Room.

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TMI Cognitive Level Rating: **Q 017**

Rating Rationale:

Answer requires associating CRD breakers/electronic trips with RPS cabinets, knowing the functions of the Fault Reset pushbutton, and applying the 2-out-of-four logic to the tripped CRD breakers/electronic trips to determine which rod groups would be deenergized.

Question:

The plant is at 100% power.

The following plant conditions exist:
- Reactor Protection System surveillance in progress on "C" RPS cabinet.
- "C" RPS cabinet's associated CRD breakers/electronic trips were tripped
- An AO has manually reclosed "C" RPS cabinets associated DC breakers
- The Fault Reset pushbutton has NOT been pushed on the CRD Diamond panel

If a loss of Vital Bus "D" were to occur at this time, which ONE of the following would be the immediate effect?

A. CRD Groups 1 and 3 would drop

B. CRD Groups 1, 2, 3 and 4 would drop

C. CRD Groups 5, 6 and 7 would drop

D. CRD Groups 1 through 7 would drop

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TMI Cognitive Level Rating: **Q 018**

Rating Rationale:

Question requires evaluation of current conditions and expected plant response.

Question:

A reactor trip occurred during performance of maintenance on the RPS cabinets. Performance of ATP 1210-1 is in progress and the following plant conditions exist:

- RCS pressure is 1800 psig and decreasing at 50 psig every 2 minutes
- OTSG A SU level is 40 inches and decreasing at 2 inches per minute
- OTSG B OP level is 95% and rising at 5% per minute
- Makeup flow is 350 gpm
- Pressurizer level is 80 inches and dropping 3 inches per minute
- SCM is 32 degrees F and dropping 1 degree per minute
- MUT level is 78 inches and dropping 10 inches per minute

Which ONE of the following identifies the most immediate operator action required for these conditions?

A. Trip all RCPs

B. Trip both MFW pumps

C. Initiate both trains of HPI

D. Open MU-V-14A or MU-V-14B, BWST suctions to the MUPs

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TMI Cognitive Level Rating: **Q 019**

Rating Rationale:

Question requires analysis to determine plant conditions under which rates of change determined by ICS are selected, and determine what the final load value will be.

Question:

Initial conditions:
- Plant at 80% power during a power escalation.
- Rod index is 275
- FW-P-1A and FW-P-1B in operation

The following sequence of events occurred:
- Loss of FW-P-1A
- Runback initiated in response to the loss of FW-P-1A
- 3 seconds after the runback initiated, a rod in Group 5 drops to the bottom of the core.

Which ONE (1) of the following describes the plant response to these conditions?

A. Runback continues at 30% per minute to 482 MWd

B. Runback continues at 30% per minute to 585 MWd

C. Runback continues at 50% per minute to 482 MWd

D. Runback continues at 50% per minute to 585 MWd

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 021**

Rating Rationale:

Application of steam tables

Question:

The following plant conditions exist:
- Small Break LOCA cooldown in progress
- cooldown rate is 30°F per hour
- incore thermocouple temperature is 456°F

ATP 1210-6, Small Break LOCA Cooldown, requires you to "Maintain OTSG pressure so that secondary T-sat is 40 to 60°F lower than incore thermocouple temperature."

Which ONE (1) of the following would be the correct OTSG pressure to accomplish this step?

A. 255 psig

B. 295 psig

C. 435 psig

D. 685 psig

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TMI Cognitive Level Rating: **Q 022**

Rating Rationale:

Requires analysis of plant data and a decision between several possible flowpaths.

Question:

A Small Break LOCA Cooldown is in progress in accordance with ATP 1210-6.

The following plant conditions exist:
- all four (4) Reactor Coolant Pumps are secured
- RCS pressure is 385 psig and stable
- both OTSG levels are 63% and being raised to 75 - 85%
- both OTSG pressures are approximately 145 psig and stable
- both T-hots are approximately 440°F
- both T-colds are approximately 400°F
- the five highest incore thermocouples average 490°F

Which ONE (1) of the following procedure transitions should take place?

A. Go To ATP 1210-4, Lack of Primary to Secondary Heat Transfer.

B. Go To ATP 1210-7, Large Break LOCA Cooldown.

C. Go To ATP 1210-8, RCS Superheated.

D. Go To ATP 1210-9, HPI Cooling - Recovery from Solid Operations.

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TMI Cognitive Level Rating: **Q 023**

Rating Rationale:

Examinee must understand the proper DHR flowpath and evaluate the effect each of the closed valves would have on that flowpath. Examinee must determine that one of the closed valves would cause pump cavitation leading to fluctuating amps.

Question:

Current plant conditions:
- The plant is in Refueling Shutdown
- "A" train Decay Heat Removal is in service
- The Reactor Coolant System is being drained down for RV Head removal
- The "B" train Decay Heat Removal is being lined up for operation

Which ONE of the following would be a reason for DH-P-1B developing fluctuating motor amps when it is started?

A. DH-V-4B closed

B. DH-V-5B closed

C. DH-V-12B closed

D. DH-V-38B closed

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TMI Cognitive Level Rating: **Q 024**

Rating Rationale:

Examinee is required to evaluate signal sources for plant logic circuits to discriminate between indications of full or partial actuations.

Question:

Plant conditions:
- ATWS in progress
- The primary CRO depresses the Manual Reactor Trip pushbutton.

Which ONE (1) of the following sets of conditions would positively ensure that the operator's action was successful AND that ALL CRDMs had power removed?

A. RPS Breaker Trip Lights on ALL 4 RPS Cabinets are bright.

B. RPS Protective Subsystem Lights on ALL 4 RPS Cabinets are bright.

C. All IN-LIMIT lamps on the Diamond Control Panel for Groups 1-7 drop are lit.

D. Alarms G-1-3 (CRD DC Trip Brkr Open) AND G-1-4 (CRD AC Trip Brkr Open) actuate.

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TMI Cognitive Level Rating: **Q 027**

Rating Rationale:

Question:

The crew has taken action to minimize subcooling margin in accordance with ATP 1210-5, OTSG Tube Leakage. Identify the ONE (1) statement below that describes the reason for minimizing subcooling margin in this situation.

A. Minimize the differential pressure between the OTSG and RCS.

B. Minimize time required for cooldown of the RCS.

C. Minimize potential of lifting Main Steam safety valves.

D. Minimize tensile stresses on affected OTSG tubes.

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TMI Cognitive Level Rating: **Q 028**

Rating Rationale:

The examinee needs to apply the steam tables to determine that the plant conditions constitute a loss of 25°F subcooling margin.

Question:

The following plant conditions exist:
- Cooling down after a "A" Steam Generator Tube Leak
- RCS pressure 985 psig
- RCS temperature 530°F
- All four Reactor Coolant Pumps operating
- 75°F/hour cooldown rate

Which ONE of the following actions should be taken:

A. Reduce cooldown rate to 50°F/hour or less

B. Increase cooldown rate toward 240°F/hour

C. Secure one Reactor Coolant Pump in each loop

D. Secure all four Reactor Coolant Pumps

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TMI Cognitive Level Rating: **Q 029**

Rating Rationale:

Requires analysis of plant operational data and application of procedural requirements to develop a planned response.

Question:

Plant conditions:
- Time is ten minutes after reactor trip due to loss of both Main Feedwater Pumps.
- EF-P-2A is operating, EF-P-1 and EF-P-2B are not operating.
- T-hot is 585° F and slowly increasing.
- RCS pressure is 2300 psig and slowly increasing.
- All RCPs are operating.
- OTSG 1A level is 25 inches.
- OTSG 1B level is 0 inches.
- OTSG 1A pressure is stable at 1010 psig.
- OTSG 1B pressure is 800 psig and decreasing.
- RCS heat up rate is +75° F/Hr.

From the list below, identify the ONE (1) required action concerning operation of the Reactor Coolant Pumps.

A. Stop 1 RCP per loop.

B. Stop 3 RCPs.

C. Stop 4 RCPs.

D. Continue to operate 4 RCPs.

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TMI Cognitive Level Rating: **Q 030**

Rating Rationale:

Requires evaluation of the given alarms and their combination to determine a common cause to arrive at the proper answer.

Question:

The following alarms, among others, are received simultaneously:
- A-1-7 Battery 1A Discharging
- A-2-7 Batt Charger 1A/1C/1E Trouble
- A-3-7 Inverter 1A/1C/1E System Trouble
- PRF1-1-1 CRDM Breaker Test Trouble
- NN-3-1 230 KV Substation Trouble
- AA-3-2 7KV Bus Trouble
- AA-3-3 4KV BOP Bus Trouble
- AA-3-5 480V BOP Bus Trouble

Which ONE (1) of the following events would result in ALL of the above indications?

A. Loss of "A" DC Distribution

B. Loss of the A/C/E Inverters

C. Loss of "A" Battery Charger

D. Loss of "A" Aux Transformer

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TMI Cognitive Level Rating: **Q 034**

Rating Rationale:

This question requires analysis of plant data, comparison of trends, prediction of how each failure would affect the plant and interpretation of discrepant data.

Question:

The plant is at 100% power with ICS in full automatic operation.

The following plant conditions exist:
- Pressurizer level 200" and decreasing about 4" per minute
- Makeup Tank level 80" and decreasing about 2" per minute
- Makeup flow 60 gpm and increasing slowly
- MU-V-17 demand at 50% and increasing slowly

Which ONE of the following failures would explain the conditions above?

A. MU-V-5 is failing open

B. MU-V-17 is failing open

C. Seal Injection flow instrument is failing low

D. Pressurizer level instrument is failing low

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TMI Cognitive Level Rating: **Q 037**

Rating Rationale:

Requires the ability to construct and interpret a 1/M plot.

Question:

The following data is obtained during an approach to criticality:

CRD Grp 5 Position	NI-11 Reading
0%	2.0 cps
25%	2.5 cps
50%	4.0 cps
75%	5.0 cps

Which ONE (1) of the following control rod positions is closest to where you would predict criticality?

A. 15% on Group 6

B. 40% on Group 6

C. 50% on Group 6

D. 95% on Group 6

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TMI Cognitive Level Rating: **Q 038**

Rating Rationale:

Requires application of given data, determining proper values from graphs and calculations to arrive at the correct answer.

Question:

Calculate an Estimated Critical Rod Position (ECP) for the following plant conditions:
- 20 hours post trip
- Plant has been running at 100% power since initial cycle startup
- 400 EFPD
- 890 ppm Boron from RCS sample
- Boron correction factor of 0.98
- CRD Group 8 at 30%
- T-ave 532°F

Which ONE (1) of the following rod tolerance bands is the most correct for this ECP?

	Minimum	Maximum
A.	72% on Group 5	55% on Group 7
B.	11% on Group 6	73% on Group 7
C.	29% on Group 6	47% on Group 7
D.	30% on Group 7	100% on Group 7

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TMI Cognitive Level Rating: **Q 039**

Rating Rationale:

Question requires evaluation of the effect a change in RCS flow conditions has on heat transfer and OTSG BTU limits, and how the ICS system will respond to that condition.

Question:

The following plant conditions exist:
- Reactor power is 70%
- 4 RCPs in operation
- ICS is in full AUTO
- High vibrations exist on RC-P-1B

Which ONE of the following describes the correct response of the feedwater system when RC-P-1B is secured?

A. Feed flow to OTSG A increases
Feed flow to OTSG B increases

B. Feed flow to OTSG A increases
Feed flow to OTSG B decreases

C. Feed flow to OTSG A decreases
Feed flow to OTSG B decreases

D. Feed flow to OTSG A decreases
Feed flow to OTSG B increases

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 041**

Rating Rationale:

Question requires evaluation of the effect that a loss of ATA will have on the ICS system.

Question:

Initial Conditions:
- Plant is at 100% power
- All ICS H/A stations are in AUTO

Which ONE (1) of the following condition(s) would exist on the loss of bus ATA?

A. MU-V-5 controller fails to the mid position

B. FW-V-17A/B controller fails to mid position.

C. Pressurizer heater control fails to 50% demand.

D. MS-V-4A/B control transfers to the Back-up Loaders.

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TMI Cognitive Level Rating: **Q 042**

Rating Rationale:

Question requires analysis of plant response and determination of cause.

Question:

The following conditions exist:
- Reactor is at 10e-8 amps and stable
- RCS boron concentration is at 950 ppm
- 'A' RCBT boron concentration is at 12 ppm
- 'B' RCBT boron concentration is at 945 ppm
- 'C' RCBT boron concentration is at 1400 ppm

Makeup is initiated to raise level in MU-T-1. Two minutes later, the following indications are observed:
- NI-3 and NI-4 indicate a 0.1 DPM startup rate
- NI-3 and NI-4 level indication is rising

Which ONE (1) of the following would result in the above plant response?

A. Makeup from the BWST.

B. Makeup from the 'A' RCBT

C. Makeup from the 'B' RCBT

D. Makeup from the 'C' RCBT

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TMI Cognitive Level Rating: **Q 043**

Rating Rationale:

Requires application of the knowledge of which containment isolation valves this actuation will close to the which components that this will affect.

Question:

The following plant conditions exist:
- The plant is at 100% power.
- An inadvertent "B" train 30# ESAS actuation has occurred
- The "B" train 30# ESAS actuation cannot be reset

Which ONE of the following are the direct consequences of this ESAS actuation?

A. High Reactor Coolant Pump Seal temperatures
High Control Rod Drive Stator temperatures

B. High Reactor Coolant Pump Seal temperatures
Loss of Reactor Coolant Pump Seal Injection

C. High Reactor Coolant Pump Motor temperatures
High Control Rod Drive Stator temperatures

D. High Reactor Coolant Pump Motor temperatures
Loss of Reactor Coolant Pump Seal Injection

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 044**

Rating Rationale:

Examinee is required to evaluate changing plant conditions to predict effects of failures.

Question:

Plant conditions:
- Reactor power is 60%, following response to an asymmetric (dropped) rod in Group 5.
- Reactor power channel SASS mismatch is present due to the dropped rod.
- All ICS stations are in AUTO.
- NI-5 is selected for ICS control.

With no operator action, from the list below identify the ONE (1) set of events that will initially result if NI-5 output fails to 0%.

A. Control rods will NOT insert/withdraw, AND feedwater flow will DECREASE.

B. Control rods will NOT insert/withdraw, AND feedwater flow will NOT change.

C. Control rods will WITHDRAW, AND feedwater flow will DECREASE.

D. Control rods will WITHDRAW, AND feedwater flow will INCREASE.

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 045**

Rating Rationale:

Examinee is required to evaluate changing plant conditions to predict effects of manual operator actions on automatic control systems.

Question:

Plant conditions:
- Reactor power is 85% following a failure in ICS SG/Rx Master.
- ICS stations in HAND:
  - SG/Rx Master.
  - Reactor Demand.
  - BOTH FW Loop Masters.
- All other stations, including the Diamond Rod Control Panel, are in AUTO.
- Control rod index is 275%.

The reactor operator inserts Group 8 rods from 30% to 20% to adjust core power imbalance. This action adds positive reactivity to the core and causes an increase in reactor power.

From the list below identify the ONE (1) statement that describes Control Rod Group 7 response during the insertion of Group 8 rods under the conditions described above.

A. Not move because neutron error will be zero (0).

B. Not move because Group-8 is being inserted.

C. Withdraw due to a positive ICS neutron error.

D. Insert due to a negative ICS neutron error.

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TMI Cognitive Level Rating: **Q 046**

Rating Rationale:

Requires the examinee to make an assumption based on given data and use both the data and assumption to obtain a value from a graph.

Question:

The following plant conditions exist:
- a pressurizer temperature RTD develops an open circuit
- pressurizer differential pressure on the PPC (A0501) is 300"
- RCS pressure is 2155 psig and stable

Which ONE (1) of the following identifies the manner in which the temperature instrument fails AND what the actual pressurizer level is?

A. low, 120"

B. low, 208"

C. high, 120"

D. high, 208"

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 049**

Rating Rationale:

Question requires evaluating temperature/pressure effects on instrumentation in an adverse atmosphere.

Question:

A large steam leak has occurred, RB pressure is 15 psig.

From the list below, identify the ONE (1) indication in the Control Room that would indicate higher than the actual condition.

A. SCM

B. RB Sump Level

C. Pressurizer Level

D. RCS Pressure

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 050**

Rating Rationale:

Question requires evaluation of plant conditions and predicting plant response.

Question:

The following plant conditions exist:
- Reactor power 100%
- Condensate Booster pumps CO-P-2A and CO-P-2C are operating
- Feed Reg Valve demands are 52%

A shaft shear occurs on CO-P-2A.

Predict which ONE (1) of the following will be the INITIAL response to this condition.

A. CO-P-2B starts automatically.

B. FW to Rx Cross Limit will occur.

C. Feed Reg Valve demands will decrease.

D. Feed Pump discharge pressures increase.

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TMI Cognitive Level Rating: **Q 051**

Rating Rationale:

Requires synthesis of plant operating conditions with equipment operational interrelationships to develop a response.

Question:

Initial plant conditions:
- Reactor power is 100%.
- CO-P-1A and CO-P-1B are running, CO-P-1C is in Normal-After-Stop.
- CO-P-2A and CO-P-2B are running, CO-P-2C is in Normal-After-Stop.
- CO-P-1B breaker trips on an electrical fault.
- After a period of two (2) seconds, CO-P-1C automatically starts.

From the list below, identify the ONE (1) statement that describes the plant response for these conditions.

A. One condensate booster pump will trip, both main feed pumps remain running, with an ICS runback.

B. One condensate booster pump and one main feed pump will trip, with an ICS runback.

C. One main feed pump will trip, both condensate booster pumps remain running, with an ICS runback.

D. Both main feed pumps trip, one condensate booster pump trips and the reactor trips.

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TMI Cognitive Level Rating: **Q 052**

Rating Rationale:

Planty conditions must be evaluated to determine the resultant effect on OTSG conditions.

Question:

The following plant conditions exist:
- A loss of condenser vacuum has occurred.
- Both MFPs are tripped.
- EFW actuated
- OTSG pressures are 1005 psig and stable.
- The IMAs of 1210-1 have been completed.

Which ONE (1) of the following describes the status of the OTSGs?

Tensile stress will be:

A. higher due to steaming through the ADVs.

B. higher due to the loss of MFW.

C. lower due to steaming through the ADVs.

D. lower due to the loss of MFW.

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TMI Cognitive Level Rating: **Q 053**

Rating Rationale:

Examinee has to relate other plant parameters to an equivalent Megawatts Electrical output to arrive at the correct answer.

Question:

A plant power reduction is being performed in order to secure the "B" Main Feedwater Pump due to high vibration.

Which ONE (1) of the following conditions would represent the HIGHEST power level acceptable for securing FW-P-1A WITHOUT exceeding an ICS High Load Limit?

A. Unit Load Demand at 65%

B. Unit Load Demand at 45%

C. Main Feedwater flow at 7.8 Mlbm/hr

D. Main Feedwater flow at 6.2 Mlbm/hr

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 054**

Rating Rationale:

Examinee must diagnose the loss of all four Reactor Coolant Pumps, diagnose the failure to swap to the proper level range and then determine how to compensate for the failure.

Question:

The following sequence of events occurred:
- there was simultaneous loss of ICCW and the running Makeup pump
- the Reactor tripped automatically
- all three Emergency Feedwater pumps started
- EFW flow to "A" OTSG is 0 gpm
- EFW flow to "B" OTSG is 400 gpm

The following status indications are present above the EF-V-30 controllers for the "A" OTSG:
- AUTO light is illuminated
- SU light is illuminated
- OP light is not illuminated

Which ONE (1) of the following is the ONLY method which will allow the operator to control the correct level in the "A" OTSG?

A. Place EF-V-30A and/or D controllers in Local Auto and dial in a 25% setpoint.

B. Place EF-V-30A and/or D controllers in Local Auto and dial in a 50% setpoint.

C. Place EF-V-30A and/or D controllers in Manual and control level manually at 25".

D. Place EF-V-30A and/or D controllers in Manual and control level manually at 50%.

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 055**

Rating Rationale:

Requires analysis of plant conditions, including discrepent information, for the examinee to arrive at the correct answer.

Question:

The following plant conditions exist:
- Fifteen (15) minutes after a reactor trip from 100% power
- All four (4) Reactor Coolant Pumps are operating
- Emergency Feedwater was actuated on low OTSG levels
- "B" OTSG level is 55" and rising at 7" per minute
- EFW flow to "B" OTSG is 300 gpm and stable
- EF-V-30B and EF-V-30C controllers in the Control Room indicate 0 demand
- An AO reports that EF-V-30B is closed and EF-V-30C is 30% open

Which ONE (1) action below is required to gain control of "B" OTSG level?

A. Secure EF-P-2A from the Control Room.

B. Order the AO to manually close EF-V-52C.

C. Order the AO to manually close EF-V-30C with the handwheel.

D. Take manual control of EF-V-30C from the Control Room positioner.

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 057**

Rating Rationale:

Examinee is required to apply knowledge of piping system interconnections to area monitoring systems sensitive to radiation in order to predict proper automatic action to mitigate consequences of an accidental radioactive gas leak.

Question:

Waste Gas Decay Tank relief valve, WDG-V-36, has opened due to high tank pressure. This valve is now FAILED OPEN.

All the radiation monitors in the release pathway go into HIGH alarm.

From the list below, identify the ONE (1) statement that describes automatic action(s) initiated by the Radiation Monitoring system related to this accidental gaseous release.

A. Trips AH-E-14A/C (B/D).

B. Trips AH-E-10 AND AH-E-11.

C. Trips AH-E-10 ONLY.

D. Trips AH-E-11 ONLY.

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TMI Cognitive Level Rating: **Q 061**

Rating Rationale:

Requires analysis of sequence of events to determine the status of the ESAS system and expected component response.

Question:

The plant was at 100% power in normal alignment when the following sequence of events occurred:
- small break LOCA
- RB pressure rose to 10 psig and stabilized
- 1600#, 500# RCS pressure ESAS signals were bypassed
- 4# RB pressure ESAS signal was defeated
- the LOCA degraded to a large break
- RB pressure rose to 40 psig

Which ONE (1) of the following identifies the equipment operating in order to limit containment pressure and temperature?

A. All three AH-E-1s in FAST speed.

B. All three AH-E-1s in SLOW speed.

C. All three AH-E-1s in FAST speed, AND both BS-P-1s.

D. All three AH-E-1s in SLOW speed, AND both BS-P-1s.

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 062**

Rating Rationale:

Requires analysis of plant data, determination of cause, and prediction that pressurizer heaters could not overcome this pressure decrease.

Question:

The following plant conditions currently exist:
- RCS pressure is 2125 psig and decreasing at 21 psig per minute
- Pressurizer temperature is 644°F and decreasing at 2.5°F per minute
- The Pressurizer Spray Valve (RC-V-1) has NO indication
- The PORV (RC-RV-2) has a green demand indication
- There is no evidence of RCS leakage.
- No operator actions have been taken.

Which ONE (1) of the following automatic OR manual actions is REQUIRED to prevent a low RCS pressure reactor trip?

A. Pressurizer Heater Bank 4 energizing at 2120 psig will mitigate the event.

B. Pressurizer Heater Bank 5 energizing at 2105 psig will mitigate the event.

C. The operator MUST close the Pressurizer Spray Block Valve (RC-V-3).

D. The operator MUST close the PORV Block Valve (RC-V-2).

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 065**

Rating Rationale:

Requires evaluation of power level in relation to nuclear instrumentation strings and associated interlocks.

Question:

During an approach to criticality, the following plant conditions exist:
- NI-11 and 12 indicate 150 cps
- NI-3 and 4 indicate 1x10-11 amps

Which ONE (1) of the following would occur if NI-4 Startup Rate fails high while the operator is withdrawing rods?

A. Rod out motion would stop immediately.

B. An out-inhibit would go into effect when NI-4 comes on scale.

C. An out-inhibit would go into effect when the operator stops withdrawing rods.

D. Rod out motion would not be affected unless either NI-11 or NI-12 indicated a high startup rate.

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 068**

Rating Rationale:

Examinee must evaluate plant response and determine corrective action.

Question:

A RB purge was just initiated using AH-E-6A and AH-E-7A when AH-E-6A trips, resulting in the purge being secured automatically.

Which ONE (1) of the following actions is required in order to restore the purge?

Start AH-E-6B, then:

A. open AH-V-1C/D.

B. open AH-V-1A/B.

C. open AH-V-1A/B AND restart AH-E-7A.

D. open AH-V-1C/D AND restart AH-E-7A.

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 070**

Rating Rationale:

Question requires evaluation of plant conditions following failure of plant equipment to operate.

Question:

The plant is at 40% power when the following events occur:
- A main steam line break occurs in the turbine building.
- The reactor is successfully tripped.
- Attempts to trip the turbine from the control room fail.
- Actions to isolate both OTSGs are initiated.
- MS-V-1A and MS-V-1C fail to close.
- All other OTSG isolation efforts are successful.

Which ONE (1) of the following conditions will occur?

A. OTSG 'A' blows down completely. OTSG 'B' is isolated.

B. OTSG 'B' blows down completely. OTSG 'A' is isolated.

C. OTSGs 'A' and 'B' blow down until the turbine is tripped locally.

D. OTSGs 'A' and 'B' blow down until generator output breakers trip on reverse power.

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: 2 **Q 072**

Rating Rationale:

Examinee is required to evaluate plant conditions to determine safety-related equipment operability under abnormal operating conditions.

Question:

Plant conditions:
- Reactor tripped from 100% power.
- LOSS OF STATION POWER has occurred.
- Primary and secondary parameters are being maintained in the P-T Plot post trip stability windows.
- NO increase in RCS activity following trip.
- Mass balance calculations indicate the existence of 5 gpm RCS leakage.
- 1210-1 Immediate Actions are being performed.

Considering the conditions above, identify the ONE (1) instrument below that, if evaluated alone, would support a valid determination that OTSG tube leakage exists.

A. RM-A-2 (RB Atmospheric Monitor).

B. RM-A-5 (Condenser Off-Gas Monitor).

C. RM-G-25 (Condenser Off-Gas Monitor).

D. RM-G-26 (Main Steam Line Monitor).

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TMI Cognitive Level Rating: **Q 074**

Rating Rationale:

Examinee is required to recognize the operational impact of plant control electrical system malfunctions, and to formulate corrective actions outside of normal procedures.

Question:

Initial plant conditions:
- Reactor power was 100%.

Sequence of events:
- Loss of entire 1A 125-250 Volt DC Electrical Distribution System.
- Manual Reactor Trip due to Main Feedwater control problems.
- Loss of RCS Subcooled Margin.

From the list below, identify the ONE (1) statement that describes how to secure the RCPs under these conditions.

A. Trip ALL 4 RCPs using their associated extension controls (4) on Console Center.

B. Trip RC-P-1A and RC-P-1C using extension controls on Console Center; Stop RC-P-1B and RC-P-1D by de-energizing 1B 6900 Volt Buses at Panel PR.

C. Trip RC-P-1B and RC-P-1D using extension controls on Console Center; Stop RC-P-1A and RC-P-1C by de-energizing 1A 6900 Volt Buses at Panel PR.

D. Stop ALL 4 RCPs by de-energizing 1A and 1B 6900 Volt Buses.

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 075**

Rating Rationale:

Question requires evaluation of effect a change in bus voltage willl have on EDG operation when paralleled with an alternate source and identification of the required corrective action.

Question:

EG-Y-1B is in parallel operation with the 1E 4160V bus for a surveillance with the following conditions:
- Exciter switch is in Manual
- 1E Bus Voltage is 4165V
- 1E Bus Frequency is 60 HZ
- Diesel load is 2.9MW
- MVARs is 2.2 MVAR

A perturbation on the 8 bus results in the 1E bus voltage reducing to 4120V.

Which ONE of the following operator actions would be required to compensate for this event?

A. Raise diesel speed using EG-Y-1B Governor Control Switch

B. Lower diesel speed using EG-Y-1B Governor Control Switch

C. Raise diesel output voltage using the Manual Voltage Controller

D. Lower diesel output voltage using the Manual Voltage Controller

NRC CRO Licensing Examination
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June 2000

TMI Cognitive Level Rating: **Q 078**

Rating Rationale:

Question requires evaluation of location of leak and plant response with failed equipment.

Question:

The following plant conditions exist:
- A rupture occurs in the air header supplying MS-V-3s.
- IA-P-4 trips on overload.
- IA-P1A/B and SA-P-1A/B are started successfully.
- Instrument air primary (PI-222) and secondary (PI-1403) pressure indicators on PL are tracking together and continue to decrease.

Which ONE(1) of the following indications would be observed on PL when IA-V-26 closes?

When header pressure drops below:

A. 60 psig, PI-222 starts to increase and PI-1403 continues to decrease.

B. 60 psig, PI-1403 starts to increase and PI-222 continues to decrease.

C. 80 psig, PI-222 starts to increase and PI-1403 continues to decrease.

D. 80 psig, PI-1403 starts to increase and PI-222 continues to decrease.

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 081**

Rating Rationale:

Examinee must evaluate the effects of each answer to to determine with would cause flow to the RCDT.

Question:

The plant is at 100% power with normal system alignments.

Which ONE (1) of the following would result in a discharge of Reactor Coolant to the Reactor Coolant Drain Tank?

A. "A" Loop RCS Wide Range T-cold failing low.

B. Leakage through RV Head vent valves RC-V-42 & 43.

C. Selected RCS Wide Range Pressure failing high without SASS actuation.

D. Closure of RCP seal return valves MU-V-25 & 26 on an inadvertant 30# ESAS.

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TMI Cognitive Level Rating: **Q 084**

Rating Rationale:

Examinee must predict component response to a partial loss of a system, and demonstrate knowledge of backup means of operation.

Question:

Turbine Bypass Valve (TBV) H/A stations are in hand with demand set at approximately 20% open. A complete loss of the normal Secondary side Instrument Air header occurs.

Which ONE (1) of the following describes the response of the TBVs and the method to control the valves?

The valves will:

A. Open. Controlled locally using the handwheel.

B. Close. Controlled locally using the handwheel.

C. Remain 20% open. Controlled from the H/A station in the CR.

D. Go to 50% open. Controlled by adjusting local air pressure to the controller.

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TMI Cognitive Level Rating: **Q 085**

Rating Rationale:

Examinee must calculate the mismatch between Reactor power and Turbine load, then compare that to the capacity of the Turbine Bypass Valves to arrive at the correct answer.

Question:

The plant has experienced a load rejection.

The following plant conditions exist:
- Turbine load is 50 Mwe
- The Diamond is in manual
- Reactor power has been stabilized at 20%

Which ONE (1) of the following design features will compensate for the mismatch between Reactor power and Turbine load?

A. The Turbine Bypass Valves will control Turbine Header pressure at 895 psig (setpoint + 10 psig bias).

B. The Turbine Bypass Valves will control Turbine Header pressure at 960 psig (setpoint + 75 psig bias).

C. The Atmospheric Dump Valves will control OTSG pressure between 1026 psig and 1052 psig.

D. The Main Steam Safety Valves will continue to lift to maintain OTSG pressure below 1040 psig.

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TMI Cognitive Level Rating: **Q 086**

Rating Rationale:

Question requires evaluating partial loss of air system effects on the Reactor River Water system and means of plant control following the loss.

Question:

RB emergency cooling was placed in operation in response to a steam leak inside containment. The air supply line to RR-V-6 fails resulting in a loss of all air to RR-V-6.

Which ONE (1) of the following actions will be required to maintain proper flow and header pressure in the Reactor River Water system?

A. Open RR-V-6.

B. Throttle RR-V-5.

C. Close RR-V-6 and throttle RR-V-5.

D. Open RR-V-6 and throttle RR-V-5.

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: **Q 087**

Rating Rationale:

Examinee has to evaluate plant conditions to determine the requirements to re-enable ESAS compnent control.

Question:

The plant was operating at 100% power when the following sequence of events occurred:
- SBLOCA
- RB pressure rose to 5 psig
- RCS pressure dropped to 400 psig
- All systems responded in automatic as required
- Manual initiation of ESAS was NOT performed

Following completion of the IMAs for the appropriate ATOG procedures, the following plant conditions are observed:
- RB pressure stabilized at 1 psig
- RCS pressure is 950 psig and rising slowly at 20 psig every 5 minutes

In order to regain control of ESAS actuated components, IAW OP 1105-3, Safeguards Actuation System, for each train the operator must push:

A. ALL 3 4 psig Enable Reset push buttons.
ALL 3 LPI Enable & Channel Reset push buttons
ALL 3 HPI Bypass push buttons

B. ALL 3 4 psig Enable Reset push buttons.
ALL 3 LPI Bypass push buttons
ALL 3 HPI Bypass push buttons

C. 2 of 3 4 psig Defeat push buttons
ALL 3 LPI Enable & Channel Reset push buttons
ALL 3 HPI Bypass push buttons

D. 2 of 3 4 psig Defeat push buttons
ALL 3 LPI Bypass push buttons
ALL 3 HPI Bypass push buttons

NRC CRO Licensing Examination
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TMI Cognitive Level Rating: 3 **Q 089**

Rating Rationale:

Examinee is required to analyze plant performance data to develop a planned response to prevent exceeding operating limits..

Question:

Plant conditions:
- Maximum Generation Emergency declared.
- Reactor power is 100%.
- Generator MVAR loading is 460 MVARS OUT.
- Main Generator Hydrogen pressure is 60 psig.

Refer to the attached Figure B-2B from OP 1106-1. From the list of values provided below, identify the MAXIMUM GENERATOR MEGAWATT OUTPUT CAPABILITY for these conditions.

A. 0 MW.

B. 600 MW.

C. 800 MW.

D. 870 MW.

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TMI Cognitive Level Rating: 2 **Q 096**

Rating Rationale:

Examinee is required to analyze radiation exposure data and exposure times under different conditions to determine the least exposure in accordance with ALARA concepts.

Question:

A job is to be performed in an area that has a general radiation dose rate of 60 mRem/hr. Installation of temporary shielding to reduce the dose rate to 10 mRem/hr will require a total dose of 30 mRem (for installation and removal).

From the list below, identify the ONE (1) option that will meet ALARA expectations for the entire job process.

A. 2 workers each taking 35 minutes to perform the job.

B. 1 worker taking 60 minutes to perform the job.

C. Utilize temporary shielding, 2 workers each taking 35 minutes to perform the job.

D. Utilize temporary shielding, 1 worker takes 60 minutes to perform the job.

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TMI Cognitive Level Rating: **Q 098**

Rating Rationale:

Question requires evaluation of the event and determination of correct response to location of the leak.

Question:

The following plant conditions exist:
- Plant power is 90% and rising at 1% per minute.
- RCS pressure is 2100 psig and dropping 5 psig per minute
- Average RCS temperature is 578 degrees F and dropping at 1 degree per minute
- MU-V-17 is opening to maintain pressurizer level at 220 inches
- Generator load is 760 MWE and decreasing at 15 MWE per minute
- Alarm B-2-8, RB Clrs Excess Condensate, is in
- RB pressure is 0.5 psig and rising at 0.1 psig per minute

Which ONE (1) of the following is the required response to these conditions?

A. Trip the reactor and go to 1210-1.

B. Reduce power to <45% and trip the turbine.

C. Commence a one hour RCS leak rate calculation.

D. Commence plant shutdown at the rate specified by the SM.

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TMI Cognitive Level Rating: **Q 099**

Rating Rationale:

Examinee has to evaluate and prioritze plant conditions to determine correct procedure entry point.

Question:

The plant was operating at 100% power when the following sequence of events occurred:
- 3 gpm OTSG tube leak is detected.
- Prior to implementing the ATOG procedures, a major steam line rupture occurs in the RB.
- Current RB pressure is 5 psig and rising at 2 psig per minute.
- All systems operated as expected for this condition.

Which ONE (1) of the following identifies the correct procedure entry for current plant conditions AND the basis for entry?

A. 1210-1, to address the core cooling upset condition.

B. 1210-3, to address the over-cooling event.

C. 1210-5, to address challenge to RCS integrity.

D. 1210-6, to address the Small Break LOCA condition.

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TMI Cognitive Level Rating: **Q 100**

Rating Rationale:

Question requires evaluation of plant conditions against requirements for throttling HPI flow.

Question:

The plant was operating at 100% power with all systems in a normal line-up when the following events occurred:
- Small break LOCA requiring reactor trip and initiation of HPI
- SCM was lost and is not being recovered
- MU-P-1B tripped when HPI was actuated

The following plant conditions exist:
- RCS pressure is 385 psig and dropping at 10 psig per minute.
- Seal injection is being maintained at 34 gpm.
- Train 'A' HPI flow is 510 gpm
- Train 'B' HPI flow is 510 gpm

Based on current plant conditions, HPI flow should be:

A. Throttled using MU-V-16A and MU-V-16B on Train 'A'.

B. Throttled using MU-V-16C and MU-V-16D on Train 'B'.

C. Maintained as is until Low Pressure injection is adequate.

D. Maintained as is until subcooling margin is restored.

Tab- Authorized Open References

Authorized Open References

Nuclear Regulatory Commission
CRO Licensing Examination
Three Mile Island Nuclear Station
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Q #	Authorized Open Reference	Justification
007	Steam Tables	TMI Obj. # V.E.03.02 Explain why it is desirable to minimize heatup and repressurization following an excessive primary to secondary heat transfer event.
014	Steam Tables	TMI Obj. # V.E.01.08 Given the plant conditions determine which IF/THEN step(s) of ATP 1210-1 apply and which other procedure(s) will be used in conjunction with the ATP 1210-1.
022	Steam Tables	TMI Obj. # V.E.08.09 Given plant conditions determine if the RCS is superheated and what actions are required.
028	Steam Tables	TMI Obj. # V.E.01.08 Given the plant conditions determine which IF/THEN step(s) of ATP 1210-1 apply and which other procedure(s) will be used in conjunction with the ATP 1210-1.
029	Steam Tables.	TMI Obj. # V.E.04.07 Given plant conditions determine if entry is required into ATP 1210-4, Lack of Heat Transfer, and state the required immediate actions.
037	OP 1103-8, Approach to Criticality, Rev. 46 pages 12 & 13	TMI Obj. # V.I.07.04 For the 1/M plot: c. Given plant data, construct and interpret the plot.
038	1103-15B, Estimated Critical Conditions, Rev 29 entire procedure Calculator Ruler	TMI Obj. # V.B.11.05 Prior to startup, perform independent estimated critical rod position and estimated critical boron concentration calculations in accordance with OP 1103-15B.

Authorized Open References

Nuclear Regulatory Commission
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Q #	Authorized Open Reference	Justification
046	1202-29, Pressurizer System Failures, Rev. 54 page 15	TMI Obj. # IV.E.01.02/V.D. Describe the failure modes of the following instruments: a. RTD Determine actual PZR level by converting the detector D/P output to level using Figure 1 of EP 1202-29.
062	Steam Tables	TMI Obj. # V.D.11.03 Given a list of symptoms, identify the event, determine what Immediate Automatic Actions will occur and what Immediate Manual Actions are required IAW EP 1202-29.
089	OP 1106-1, Turbine Generator, Rev. 94, Figure B-2B (Rev. 94), Page 59.	TMI Obj. # IV.C.09.20 Explain the relationship between main generator H2 pressure and allowable generator electrical load.

Tab - KA Ratings

NRC Question KA Ratings

Q #	Rating RO	Rating SRO	KA Item	KA Catalog Page #	Measurement
001	3.4	4.1	APE 005 AK3.04	4.2-6	Knowledge of the reasons for the following responses as they apply to the Inoperable/Stuck Control Rod: Tech-Spec limits for inoperable rods
002	3.4	3.4	APE 015/017 AA1.13	4.2-11	Ability to operate and/or monitor the following as they apply to Reactor Coolant Pump Malfunctions: Reactor power level indicators
003	3.7	4.0	BW E09 EK1.2	4.3-15	Knowledge of the operational implications of the following concepts as they apply to Natural Circulation Cooldown: Normal, abnormal and emergency operating procedures associated with Natural Circulation Cooldown
004	3.6	3.9	APE 024 AK1.02	4.2-14	Knowledge of the operational implications of the following concepts as they apply to Emergency Boration: Relationship between boron addition and reactor power
005	2.7	2.8	APE 026 AA1.04	4.2-18	Ability to operate and/or monitor the following as they apply to the Loss of Component Cooling Water: CRDM high-temperature alarm system
006	3.3	3.2	APE 027 AA1.05	4.2-20	Ability to operate and/or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: Transfer of heaters to backup power supply
007	4.2	3.8	BW E05 EK3.3	4.3-10	Knowledge of the reasons for the following responses as they apply to the Excessive Heat Transfer: Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations
008	3.9	4.1	APE 051 AA2.02	4.2-34	Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip
009	3.3	3.4	EPE 055 GEN EP 2.4.31	2-14	Station Blackout Knowledge of annunciators alarms and indications, and use of the response instructions
010	2.9	3.6	APE 062 AA2.02	4.2-29	Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The cause of possible SWS loss
011	3.3	4.1	APE 067 AK3.04	4.2-52	Knowledge of the reasons for the following responses as they apply to the Plant Fire on Site: Actions contained in EOP for plant fire on site
012	2.9	3.1	APE 068 AK2.03	4.2-54	Knowledge of the interrelations between the Control Room Evacuation and the following: Controllers and positioners

Q #	Rating RO	Rating SRO	KA Item	KA Catalog Page #	Measurement
013	3.5	3.7	APE 069 AA1.01	4.2-57	Ability to operate and/or monitor the following as they apply to the Loss of Containment Integrity: Isolation valves, dampers, and electropneumatic devices
014	3.7	4.2	EPE 074 EA2.04	4.1-17	Ability to determine or interpret the following as they apply to a Inadequate Core Cooling: Relationship between RCS temperature and main steam pressure
015	3.2	3.8	APE 076 AK3.06	4.2-58	Knowledge of the reasons for the following responses as they apply to the High Reactor Coolant Activity: Actions contained in EOP for high reactor coolant activity
016	3.8	3.8	BW A02 AK2.2	4.3-28	Knowledge of the interrelations between the Loss of NNI-X/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
017	2.5	2.8	APE 003 AK2.05	4.2-4	Knowledge of the interrelations between the Dropped Control Rod and the following: Control rod drive power supplies and logic circuits
018	3.2	3.8	BW E02 EA2.2	4.3-3	Ability to determine and interpret the following as they apply to the VSSV/Post-Trip Stabilization: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments
019	3.7	3.7	BW A01 AA1.3	4.3-27	Ability to operate/monitor the following as they apply to the Runback: Desired operating results during abnormal/emergency situations
020	3.0	3.3	BW A04 AK1.1	4.3-32	Knowledge of the operational implications of the following concepts as they apply to the Turbine Trip: Components, capacity, and functions of emergency systems.
021	3.4	3.4	EPE 009 EA1.14	4.1-5	Ability to operate and/or monitor the following as they apply to a small break LOCA: Secondary pressure control
022	4.0	4.3	BW E08 2.4.4	2-11	LOCA Cooldown: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency/abnormal procedures
023	2.7	2.9	APE 025 AA2.01		Ability to determine and interpret the following as they apply to the Loss of RHR System: Proper amperage of running LPI/DHR pumps
024	4.5	4.5	EPE 029 EA1.08	4.1-9	Ability to operate and/or monitor the following as they apply to a ATWS: Reactor trip switch pushbutton

Q #	Rating RO	Rating SRO	KA Item	KA Catalog Page #	Measurement
025	3.2	3.6	APE 032 AK3.01	4.2-24	Knowledge of the reasons for the following responses as they apply to the Loss of SR NI: Startup termination on source-range loss
026	2.7	3.0	APE 033 AK1.01	4.2-26	Knowledge of the operational implications of the following concepts as they apply to Loss of IR NI: Effects of voltage changes on performance
027	4.2	4.4	APE 037 AK3.07		Knowledge of the reasons for the following responses as they apply to the S/G Tube Leak: Actions contained in EOP for S/G tube leak
028	4.1	4.2	EPE 038 EK3.08	4.1-11	Knowledge of the reasons for the following responses as they apply to the SGTR: Criteria for securing RCP
029	4.4	4.6	APE 054 AK3.04	4.2-35	Knowledge of the reasons for the following responses as they apply to the Loss of MFW: Actions contained in EOPs for loss of MFW
030	2.8	3.1	APE 058 AK1.01	4.2-43	Knowledge of the operational implications of the following concepts as they apply to the Loss of DC: Battery charger equipment/instrumentation
031	2.7	2.7	APE 059 AK2.02		Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following: Radioactive-gas monitors
032	3.6	3.8	APE 060 AA2.06	4.2-47	Ability to determine/interpret the following as they apply to the Accidental Gaseous Release: Valve lineup for release of radioactive gases
033	3.1	3.5	APE 061 AA2.04	4.2-48	Ability to determine and interpret the following as they apply to the Area RM System Alarms: Whether an alarm channel is functioning properly
034	3.3	3.4	APE 028 AA2.10	4.2-23	Ability to determine or interpret the following as they apply to the Pressurizer Level Control Malfunctions: Whether the automatic mode for PZR level control is functioning improperly, necessity of shift to manual modes
035	4.0	4.3	BW A08 2.4.4	2-11	Refueling Canal Level Decrease: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency/abnormal procedures
036	3.5	3.9	APE 056 AK3.01	4.2-37	Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Order and time to initiation of power for the load sequencer
037	4.1	4.4	SYSTEM 001 A1.06	3.1-8	Ability to predict/monitor changes in parameters associated with operating the CRDS controls including: Reactor power
038	3.5	3.9	SYSTEM 001 A4.10	3.1-10	Ability to manually operate and/or monitor in the control room: Determination of an ECP

Q #	Rating RO	Rating SRO	KA Item	KA Catalog Page #	Measurement
039	3.2	3.5	SYSTEM 003 K5.04	3.4-7	Knowledge of the operational implications of the following concepts as they apply to the RCPs: Effects of RCP shutdown on secondary parameters, such as steam pressure, steam flow, and feed flow
040	2.9	2.9	SYSTEM 003 A1.02	3.4-8	Ability to predict/monitor changes in parameters associated with operating the RCPs controls including: RCP pump and motor bearing temperatures
041	2.6	2.7	SYSTEM 004 K2.06	3.1-13	Knowledge of bus power supplies to the following: Control instrumentation
042	2.8	3.2	SYSTEM 004 K5.07	3.1-14	Knowledge of the operational implications of the following concepts as they apply to the CVCS: Relationship between SUR and reactivity
043	4.3	4.7	SYSTEM 013 K3.03	3.2-25	Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: Containment
044	2.9	3.2	SYSTEM 015 K3.06	3.7-5	Knowledge of the effect that a loss or malfunction of the NIS will have on the following: Reactor regulating system
045	3.4	3.7	SYSTEM 015 K4.08	3.7-6	Knowledge of NIS design features and/or interlocks provide for the following: Automatic rod motion on demand signals
046	3.9	4.2	SYSTEM 004 A2.02	3.1-18	Ability to predict the impacts of the following malfunctions or operations on the CVCS; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Loss of PZR level (failure mode)
047	3.8	4.1	SYSTEM 017 A4.02	3.7-12	Ability to manually operate and/or monitor in the control room: Temperature values used to determine RCP operation during inadequate core cooling
048	3.5	3.7	SYSTEM 022 K1.01	3.5-5	Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: SWS
049	2.9	3.2	SYSTEM 022 K3.01	3.5-5	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
050	2.6	2.6	SYSTEM 056 K1.03		Knowledge of the physical connections and/or cause-effect relationships between the Condensate System and the following systems: MFW

Q #	Rating RO	Rating SRO	KA Item	KA Catalog Page #	Measurement
051	2.6	2.8	SYSTEM 056 A2.04	3.4-39	Ability to predict the impacts of the following malfunctions or operations on the Condensate System; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Loss of condensate pumps
052	3.5	3.7	SYSTEM 059 K3.03	3.4-41	Knowledge of the effect that a loss or malfunction of the MFW will have on the following: S/Gs
053	2.7	2.9	SYSTEM 059 A1.03	3.4-43	Ability to predict/monitor changes in parameters associated with operating the MFW controls including: Power level restrictions for operation of MFW pumps and valves
054	2.5	2.8	SYSTEM 061 K6.01		Knowledge of the effect that a loss of the following will have on the AFW components: Controllers and positioners
055	3.9	3.4	SYSTEM 061 GEN. 2.1.3	2-4	Ability to locate and operate components, including local controls
056	3.3	3.3	SYSTEM 068 A2.04	3.9-4	Ability to predict the impacts of the following malfunctions or operations on the Liquid Radwaste System; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Failure of automatic isolation
057	3.6	3.8	SYSTEM 071 A3.03	3.9-7	Ability to monitor automatic operation of the Waste Gas Disposal System including: Radiation monitoring system alarm and actuating signals
058	2.5	3.2	SYSTEM 072 K5.02	3.7-13	Knowledge of the operational implications of the following concepts as they apply to the ARM system: Radiation intensity changes with source distance
059	3.9	4.1	SYSTEM 072 GEN. 2.4.2	2-11	Knowledge of system setpoints, interlocks and automatic actions associated with EOP entry conditions
060	2.5	3.7	SYSTEM 002 GEN.2.2.25	2-7	Knowledge of bases in Tech Specs for LCOs and safety limits
061	4.2	4.4	SYSTEM 006 K3.03	3.2-17	Knowledge of the effect that a loss or malfunction of the ECCS will have on the following: Containment
062	3.9	3.9	SYSTEM 010 A2.02	3.3-8	Ability to predict the impacts of the following malfunctions or operations on the PZR PCS system; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Spray valve failures
063	3.7	3.8	SYSTEM 011 K1.02	3.2-21	Knowledge of the physical connections and/or cause-effect relationships between the PZR LCS and the following systems: RCS

Q #	Rating RO	Rating SRO	KA Item	KA Catalog Page #	Measurement
064	2.9	3.4	SYSTEM 012 A1.01	3.7-3	Ability to predict/monitor changes in parameters associated with operating the RPS controls including: Trip setpoint adjustment
065	3.1	3.3	SYSTEM 014 K4.05	3.1-21	Knowledge of RPIS design features and/or interlocks provide for the following: Rod hold interlocks
066	3.2	3.2	SYSTEM 016 K1.03	3.7-9	Knowledge of the physical connections and/or cause-effect relationships between the NNIS and the following systems: Steam dump system
067	3.4	3.6	SYSTEM 026 K2.01	3.5-10	Knowledge of bus power supplies to the following: Containment spray pumps
068	2.7	3.1	SYSTEM 029 A2.03	3.8-7	Ability to predict the impacts of the following malfunctions or operations on the Containment Purge System; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Startup operations and the associated required valve lineups
069	2.9	3.1	SYSTEM 033 A3.02	3.8-10	Ability to monitor automatic operation of the Spent Fuel Pool Cooling System including: Spent fuel leak or rupture
070	3.2	3.6	SYSTEM 035 K6.01	3.4-15	Knowledge of the effect that a loss of the following will have on the S/Gs: MSIVs
071	3.1	3.5	SYSTEM 039 A3.02	3.4-21	Ability to monitor automatic operation of the MRSS including: Isolation of the MRSS
072	2.6	2.6	SYSTEM 055 K1.06	3.4-33	Knowledge of the physical connections and/or cause-effect relationships between the CARS and the following systems: Process radiation monitors
073	2.7	3.0	SYSTEM 062 A2.08	3.6-4	Ability to predict the impacts of the following malfunctions or operations on the AC Distribution System; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Consequences of exceeding voltage limitations
074	3.5	3.7	SYSTEM 063 K3.02	3.6-6	Knowledge of the effect that a loss or malfunction of the DC Electrical System will have on the following: Components using DC control power
075	3.1	3.1	SYSTEM 064 A2.03	3.6-10	Ability to predict the impacts of the following malfunctions or operations on the ED/G system; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Parallel operation of ED/Gs
076	3.3	3.9	SYSTEM 073 K4.02	3.7-15	Knowledge of PRM system design features and/or interlocks provide for the following: Letdown isolation on high RCS activity

Q #	Rating RO	Rating SRO	KA Item	KA Catalog Page #	Measurement
077	3.9	4.1	SYSTEM 075 GEN. 2.4.2	2-11	Knowledge of system setpoints, interlocks and automatic actions associated with EOP entry conditions
078	2.7	2.7	SYSTEM 079 A4.01	3.8-22	Ability to manually operate and/or monitor in the control room: Cross-tie valves with IAS
079	3.3	3.3	SYSTEM 086 A4.01	3.8-24	Ability to manually operate and/or monitor in the control room: Fire water pumps
080	3.1	2.9	SYSTEM 005 A4.04	3.4-13	Ability to manually operate and/or monitor in the control room: Controls and indication for closed cooling water pumps
081	3.0	3.2	SYSTEM 007 K1.03	3.5-2	Knowledge of the physical connections and/or cause-effect relationships between the PRTS and the following systems: RCS
082	3.1	3.4	SYSTEM 027 K5.01	3.5-14	Knowledge of the operational implications of the following concepts as they apply to the CIRS: Purpose of charcoal filters
083	2.5	3.1	SYSTEM 034 A3.02	3.8-13	Ability to monitor automatic operation of the Fuel Handling System including: Load limits
084	2.8	3.1	SYSTEM 041 A2.03	3.4-25	Ability to predict the impacts of the following malfunctions or operations on the SDS; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Loss of IAS
085	2.8	3.0	SYSTEM 045 K4.42	3.4-29	Knowledge of MT/G system design features and/or interlocks provide for the following: Operation of turbine bypass in event of load loss or plant trip
086	3.4	3.6	SYSTEM 078 K3.02	3.8-19	Knowledge of the effect that a loss or malfunction of the IAS will have on the following: Systems having pneumatic valves and controls
087	3.6	3.8	SYSTEM 103 K1.08	3.5-18	Knowledge of the physical connections and/or cause-effect relationships between the containment system and the following systems: SIS, including action of safety injection reset
088	3.0	3.4	Conduct of Operations 2.1	2-1	Knowledge of shift turnover practices.
089	2.8	3.1	Conduct of Operations 2.1	2-3	Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data.
090	3.4	3.3	Conduct of Operations 2.1	2-4	Knowledge of how to conduct and verify valve lineups: performance of valve lineup - independent verification of valve position.
091	3.4	3.8	Conduct of Operations 2.1	2-4	Ability to explain and apply all system limits and precautions.
092	2.5	3.4	Equipment Control 2.2.11	2-6	Knowledge of the process for controlling temporary changes.

Q #	Rating		KA Item	KA Catalog Page #	Measurement
	RO	SRO			
093	3.0	3.4	Equipment Control 2.2.12	2-6	Knowledge of surveillance procedures.
094	3.6	3.8	Equipment Control 2.2.13	2-6	Knowledge of tagging and clearance procedures.
095	2.6	3	Radiation Control 2.3.1	2-9	Knowledge of 10 CFR: 20 and related facility radiation control requirements.
096	2.5	2.9	Radiation Control 2.3.2	2-9	Knowledge of facility ALARA program.
097	2.5	3.1	Radiation Control 2.3.4	2-9	Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.
098	3.4	3.6	Emerg Procedures/Plan 2.		Knowledge of abnormal condition procedures.
099	2.7	3.6	Emerg Procedures/Plan 2.	2-13	Knowledge of the specific bases for EOPs.
100	4.0	4.0	Emerg Procedures/Plan 2.		Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Form ES-401-6

Written Exam Question Worksheet

Q # 001

Examination Outline Cross-Reference RO SRO

KA # APE 005 AK3.04 Page 4.2-6 Tier # 1

RO/SRO Importance Rating 3.4 4.1 Group # 1

Measurement Knowledge of the reasons for the following responses as they apply to the Inoperable/Stuck Control Rod: Tech-Spec limits for inoperable rods

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

Sequence of events:
- Plant STARTUP in progress (4 RCPs operating).
- Power escalation stopped at 69% reactor power.
  - Group 7 Rod #1 mechanically stuck, 10 inches below the rest of Group 7.
  - Group 7 Absolute Group Average position indication = 55% withdrawn.
- 90 minutes later:
  - Rod #1 is unstuck, withdrawn and re-aligned with remaining Group 7 rods.
  - Power escalation to 100% is resumed at 30% per hour.

Identify the ONE (1) statement below that describes IF and WHY the above actions are procedurally ACCEPTABLE or UNACCEPTABLE in accordance with EP 1202-8, CRD Equipment Failure.

A. Actions are ACCEPTABLE; adverse power peaking will not occur - the stuck rod was re-aligned with its group within two (2) hours.

B. Actions are ACCEPTABLE; adverse power peaking will not occur - the stuck rod was re-aligned with its group while the group was less than 60% withdrawn.

C. Actions are UNACCEPTABLE; adverse power peaking could occur - the rate of power escalation was too rapid after the stuck rod was re-aligned with its group.

D. Actions are UNACCEPTABLE; adverse power peaking could occur - the stuck rod was (withdrawn) re-aligned with Group 7 with reactor power greater than 60%.

Technical Reference EP 1202-8, CRD Equipment Failure, Rev. 53, Pages 3 through 8.

Open Exam Reference

Learning Objective V.D.03.04 Given a misaligned control rod and time since misalignment determine the steps to be taken to re-align the rod and explain the bases for those actions per EP 1202-8.

Question Source ☐ New ☑ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question # 1

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5, .10 ☐ 55.43 ☑ 55.45 .6, .13

Discriminant Validity Statements

A Plausible since EP 1202-8 actions for stuck rod allow two (2) hours to reduce reactor power below 60%.

B Plausible since EP 1202-8 allows asymmetric rod recovery if greater than one (1) hour - only if reactor power is less than 60%.

C Plausible since EP 1202-8 limits power escalation to 3% per hour following recovery from an asymmetric rod condition for greater than 24 hours.

D Correct answer.

Comments None.

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GPU NUCLEAR	TMI Emergency Procedure	Number Q-01 1202-8

Title	Revision No.
CRD Equipment Failure	53

Applicability/Scope	Responsible Office	Effective Date
TMI Division	Plant Operations Director	10/12/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator		10/7/99
Procedure Owner		10-8-99
PRG		10/9/99
Approver		10/8/99

A. **ASYMMETRIC ROD FAULT**

1A. Symptoms

1. Asymmetric rod fault light on the Diamond Rod Control Panel.

2. One or more fault lights lit on the rod position indication panel.

3. CRD pattern asymmetrical alarm on the main annunciator (G-2-1).

4. Rod mis-alignment indicated by position indication meters and the PPC.

5. Power Distribution Limits Exceeded alarm on the main Annunciator (G-2-6.)

6. PPC alarm L3039 "7 INCH ASYMMETRIC ROD".

> **NOTE**
>
> For a **DROPPED ROD**, follow Section B.

2A. Immediate Action

1. Automatic Action

   a. A CRD OUT-INHIBIT occurs if ICS selected reactor power is greater than 60% with the Diamond Panel in AUTO.

2. Manual Actions

> **NOTE**
>
> The steps marked with an asterisk (*) shall be reverified as the first step in the follow-up action.

> **CAUTION**
>
> If a rod remains misaligned for greater than one (1) hour, adverse power peaking may occur during the re-alignment which could lead to fuel damage when reactor power is greater than 60% of the power allowed for the RC-P combination.

   *a. Immediately evaluate for a PI problem or misaligned rod.

NOTE

Evaluation of possible PI Problems should include the following:

1. Check the imbalance meters on the console for flux imbalance.
2. Check NAS display #1 for evidence of tilt.
3. CRD groups 1 through 4 - verify the "Out-Limit" lamp on the PI panel for the affected rod is ON.
4. CRD Groups 5, 6, 7 or 8 - insert the affected group or rod a short distance to see if the individual PI jumps back to the group average position.
5. Have I & C check the operation of the API amplifier per the "Position Indicator Amplifier Check" section of 1430-CRD-19.

CAUTION

Realignment of the misaligned rod per the following step must occur within 1 hour of its discovery per Tech. Spec. 3.5.2.2.f. Go to Immediate Manual Action a.2 if realignment cannot be accomplished within 1 hour.

1. If it is determined that there is no PI problem with the rod, and the rod has been misaligned for less than 60 minutes, then attempt to realign the rod with the rest of the group using the follow-up actions of this procedure.

2. If the rod position cannot be determined, or if the immediate attempt to realign the rod is unsuccessful, then reduce reactor power to less than 60% of the allowable power for the RC-P combination within 60 minutes of discovery of the misaligned rod. And the overpower trip setpoint shall be reduced to ≤ 70% of the thermal power allowable within 10 hrs.

CAUTION

On an asymmetric rod without an IN-LIMIT, manual action is required to assure compliance with Tech. Specs Section 3.5.2

*b. If a runback is required, and feedwater, the reactor or the turbine is in MANUAL, then run the station in manual back to 60% of the allowable power for the RC-P combination.

NOTE

Compatible values of the neutron power, MWe and feedwater flow are found on the nomogram located on console CC.

3A. Follow Up Action

OBJECTIVE: Confirm rod position of the affected rod, and realign the rod with the group average. In case rod position cannot be confirmed, reduce reactor power to below 60% of the limit for the RC-P combination. Watch for quadrant power tilt and reduce power per Tech. Specs. to ensure core power distribution is acceptable. Applicable Tech Spec sections are 3.5.2 and the Core Operating Limits report.

____ 1. Reverify the steps in the Immediate Manual Actions that are marked with an asterisk (*). Use redundant indications where available.

2. Verify quadrant power tilt is within allowable limits per Tech. Specs.

____ a. If quadrant power tilt exceeds Tech. Spec. limits, or power/imbalance/tilt alarm MAP G-2-6 annunciates, obtain printouts for NAS displays 1, 4, 5, 18 and 20. The power reduction, if required due to tilt, should be accomplished per OP-1102-4, *Power Operations*.

> **NOTE**
>
> RPS trip setpoint adjustments that may need to be made in accordance with the following step are to be performed in accordance with 1430-RPS-3, *RPS Flux/Flow/Imbalance Setpoint Reduction*.

____ b. Within 10 hours tilt shall be reduced to less than the tilt limit (except for physics tests) or comply with the requirements of Tech. Specs. 3.5.2.4.e.1 through 3.5.2.4.e.4.

____ c. Verify compliance with Tech. Specs. 3.5.2.4.f and 3.5.2.4.g.

____ 3. Following the power reduction, verify rod overlap between rod groups 6 and 7 is 25 +/-5%.

> **NOTE**
>
> Rod group overlap is determined by the formula [100-(Grp 6 - Grp 7)] = overlap. G-2-6, L3057 and/or G-2-2 (Sequence Fault) will alarm if overlap is abnormal.

____ a. If the rod group overlap is NOT correct, select SEQUENCE OVERRIDE and INSERT group 6 or 7 to obtain the proper overlap.

____ 4. If in AUTO, obtain a steady state plant operating condition and select MAN at the Diamond Rod Control panel.

____ 5. If in MAN on the Diamond Rod Control Panel, cease rod motion unless continued insertion is required by plant conditions.

NOTE

If the ABSOLUTE and RELATIVE positions agree, but differ from the group average position, then the CRDM is NOT responding to commands. Review previous transfer operations and check for faulty transfer switch or faulty transfer logic equipment. If the RELATIVE position agrees with the group average, but the ABSOLUTE position indication is different from the group average, then suspect a faulty mechanism, blown motor output fuses or a faulty absolute position indication amplifier.

____ 6. Compare the absolute to the relative position by alternately selecting ABSOLUTE and RELATIVE with the POSITION SELECT switch.

____ 7. Reset the relative rod position indication for the affected rod as necessary using Section 3.4.1 of OP-1105-9, *Control Rod Drive System.*

____ 8. If the affected Group is on the Aux Supply, check the CONTROL ON lamp on the PI Panel. If the CONTROL ON lamp is not illuminated, the mechanism did NOT transfer with the rest of the group.

NOTE

If the affected group is at its OUT-LIMIT, the GROUP OUT-LIMIT must be cleared prior to re-aligning the rod.

____ 9. Attempt to exercise the asymmetric rod by transferring it to the Aux Supply and moving it toward the remaining rods in the group in RUN speed. Refer to Section 3.2.2 of OP-1105-9, *Control Rod Drive System.*

____ 10. Observe the phase rotation lights in the affected rod's power supply cabinet when first attempting to move the rod.

____ 11. If the rod responds normally, attempt to move the rod in the opposite direction in RUN speed.

____ 12. If the rod does not respond, refer to the **STUCK ROD** section of this procedure.

____ 13. If the rod responds normally, move the affected rod to the group average position by comparing the rod absolute position at the PI Panel to the Group average position on the console.

____ 14. Transfer the rod back to its normal power supply. Refer to Section 3.2.3 of OP-1105-9, *Control Rod Drive System.*

____ 15. Return the rod groups to their correct sequence and overlap if necessary, and resume normal operation.

_____ 16. If the rod responds to commands, but lags the remainder of the rods in the group, keep the rod within 7" of the group by stopping rod motion, and re-aligning the rod with rest of the group by placing it on the Auxiliary Power Supply and positioning it at the group average. Refer to OP-1105-9, *Control Rod Drive System*, for transfers to and from the Auxiliary Power Supply.

_____ 17. Obtain the following computer printouts at the specified intervals pending further direction from the nuclear engineers.

a.	NAS display #1	Repeat every half-hour
b.	NAS display #5	Repeat every half hour
c.	NAS display #17	Repeat every hour
d.	NAS display #18	Repeat every 2 hours

18. If the rod cannot be exercised, and it is outside the allowable average position or, if both the absolute and relative position indications are inoperable, then perform the following within one (1) hour of declaring the rod inoperable:

_____ a. Maintain reactor power at or below 60% of the allowable power for the RC-P combination.

_____ b. Exercise the remaining rods to verify operability using SP-1303-3.1, *Control Rod Movement*.

_____ c. Tech. Spec. 3.5.2 requires that an evaluation be initiated immediately to verify the existence of a 1% delta k/k hot shutdown margin.

NOTE

A shutdown margin of at least 1% must be present when the worth of an inoperable rod is combined with the highest worth operable rod.

_____ d. If within one hour of determination of an inoperable rod as defined in Tech. Spec. 4.7.1 and once per 12 hours thereafter, it is not determined that a 1% delta k/k hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the HOT SHUTDOWN condition within 6 hours until this margin is established.

_____ e. If a shutdown margin of 1% or greater is verified within one hour of the determination that a rod is inoperable, then trim the remaining rods in the group to the position of the inoperable rod and continue normal operation. Maintain the inoperable rod within the limits of the allowable group average position.

_____ 19 Continue to evaluate shutdown margin at twelve hour intervals in accordance with Tech. Spec. 3.5.2.2.c.

_____ 20. If the computer is unavailable, calculate tilt and imbalance every half hour per AP-1203-7, *Hand Calculations for Quadrant Power Tilt and Core Power Imbalance*.

_____ 21. If the asymmetric condition exists for more than 24 hours, then power escalation following realignment shall be limited to 3% power/hour.

B. **DROPPED ROD**

1B. Symptoms

1. Unexpected In-limit indication.
2. CRD pattern asymmetrical alarm (G-2-1).
3. Absolute position shows one or more rods at 0% on both individual position indication meters and the PPC.
4. Flux tilt as indicated by incore and out-of-core detectors and NAS display #1.
5. Reduction in reactor power level with accompanying fluctuations in reactor coolant temperature and pressure, and pressurizer level.
6. ICS RUNBACK alarm (H-1-1) if Rx power is greater than 60% and the Diamond Panel is in AUTO.
7. Power Distribution Limits Exceeded alarm (G-2-6).
8. PPC alarm L3039 "7 INCH ASYMMETRIC ROD".
9. Asymmetric Rod Alarm on the Diamond Rod Control Panel.
10. Out-Inhibit Alarm on the Diamond Rod Control Panel if the Asymmetric Rod Alarm is received and reactor power is greater than 60%.

2B. Immediate Action

1. Automatic Action
   a. If ICS is in full auto, the plant will run back to 482 MWe.
   b. Possible reactor trip on low RCS pressure.
   c. Low pressurizer level may result from reduced reactor power and T_{ave}.
2. Manual Actions

NOTE
If the Diamond Rod Control Panel is in manual and a runback is required due to a dropped rod, the in-limit bypass pushbutton must be pressed to insert the group which contains the dropped rod.

Form ES-401-6

Written Exam Question Worksheet

Q # 002

Examination Outline Cross-Reference

					RO	SRO
KA #	APE 015/017 AA1.13	Page	4.2-11	Tier #	1	
RO/SRO Importance Rating	3.4	3.4		Group #	1	

Measurement Ability to operate and/or monitor the following as they apply to Reactor Coolant Pump Malfunctions: Reactor power level indicators

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

RC-P-1A has been secured due to a #1 seal failure. It is desired to continue operating at 65% power.

Which ONE of the following is closest to the expected core delta-T for this power level?

A. 22.4

B. 29.9

C. 34.5

D. 39.9

Technical Reference OP 1102-2, Plant Startup, rev.134 page 6.

Open Exam Reference

Learning Objective V.I.07.12 Describe how operators could determine if a gross miscalibration of the NIs existed.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge

☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .5, .6

Discriminant Validity Statements

A Plausible answer calculated for 4 RCPs running and 75% power limit, then adjusted to 65% power.
Calculation: 46/100 = x/75 x = 34.5 x * 0.65 = 22.4

B Plausible answer calcalated for 4 RCPs running.
Calculation: 46/100 = x/65 x = 29.9

C Plausible answer calculated for 4 RCPs running and 75% power limit.
Calculation: 46/100 = x/75 x = 34.5

D Correct answer.
Ratio should be: 46°F delta-T/RCP power limit = actual delta-T/actual power
Calculation: 46/75 = x/65 x = 39.9

Comments None.

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AmerGen	TMI - Unit 1 Operating Procedure	Number 1102-2
Title **Plant Startup**		Revision No. 134
Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date 04/12/00

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	W J Marshall	4/12/00
Procedure Owner	D W Blair for J. BLAIR	4/12/00
PRG	Howard C Crawford Vice Chairman	4/12/00
Approver	D W Blair	4/12/00

16. Perform all Power Range Calibrations in accordance with 1302-1.1.

    Regular and frequent cross-checking of indicated NI power with other thermohydraulic indicators of real reactor power (i.e.: core ΔT, MWe FW flow) can provide early warning of non-conservative NI calibration and/or other process problems requiring resolution.

17. When RCS temperature is less than 525 DEGF, maintain RCS boron concentration greater than or equal to the Cold Shutdown concentration per Figure 1 of 1103-4.

    When RCS temperature is greater than or equal to 525 DEGF and OTSG level is greater than 40 inches, maintain RCS boron concentration greater than or equal to the Flooded Nozzle concentration per Figure 1 of 1103-4.

    When RCS temperature is greater than or equal to 525 DEFG and OTSG level is less than or equal to 40 inches, maintain RCS boron concentration greater than or equal to the Hot Shutdown concentration per Figure 1 of 1103-4.

    For initial cycle startups, following a refueling outage, the required boron concentration shall be determined by the Manager, Shift Engineering.

18. Manning requirements shall be in compliance with Administrative Procedure 1029.

19. A minimum of 107 KW of pressurizer heaters, from each of two pressurizer heater groups shall be operable. Each operable 107 KW of pressurizer heaters shall be capable of receiving power from a 480 volt ES bus via the established manual transfer scheme. (T.S. 3.1.3.4.2)

20. The PORV and the associated block valve shall be operable during HOT STANDBY, STARTUP, and POWER OPERATION. Refer to T.S. 3.1.12.4 for time clock limits if either valve becomes inoperable.

21. The minimum number of accident monitoring channels identified for the instruments in Table 3.5-2 of T.S., shall be operable. (Refer to T.S. 3.5.5.1).

22. At least one Reactor Coolant System vent path consisting of at least two power operated valves in series, powered from emergency buses shall be operable and closed at each of the locations listed in T.S. 3.1.13 when the reactor is critical. Refer to T.S. 3.1.13.2 for time clock limits when valves are not operable.

23. If EFW actuates while the reactor is critical and main feedwater is unavailable, shutdown the reactor. When Rx power exceeds the capacity of EFW to provide an adequate heat transfer balance, manual reactor trip should be considered.

24. If an RB purge is in progress obtain an RMA-9 sample per SP 1301-4.7 (ODCM, Table 3.2-2, Note b) as necessary.

25. Review transient cycle logging requirements prior to commencing startup.

26. If at any time the most conservative valid wide range cold leg temperature indication shows a RCS temperature less than 525°F, verify or make the reactor at least 1% ΔK/K shutdown. Insert control rods or borate as necessary.

Form ES-401-6

Written Exam Question Worksheet

Q # 003

Examination Outline Cross-Reference RO SRO

KA # BW E09 EK1.2 Page 4.3-15 Tier # 1

RO/SRO Importance Rating 3.7 4.0 Group # 1

Measurement Knowledge of the operational implications of the following concepts as they apply to Natural Circulation Cooldown: Normal, abnormal and emergency operating procedures associated with Natural Circulation Cooldown

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

A Natural Circulation Cooldown was in progress when the following sequence of events occurred:
- MS-V-4A failed open
- cooldown rate increased to 120°F/hour
- Pressurizer level increased from 100" to 250" within a few seconds
- MS-V-4A was taken under manual control
- cooldown rate was reduced to 40°F/hour

Which ONE (1) of the following are the proper actions to take following this sequence of events?

A. Stop the cooldown and maintain RCS temperature constant.
Maintain RCS pressure at the current value or slightly higher.

B. Stop the cooldown and maintain RCS temperature constant.
Continue RCS depressurization within the Natural Circulation Pressure-Temperature Limits curve.

C. Continue cooldown at less than 50°F/hour.
Maintain RCS pressure at the current value or slightly higher.

D. Continue cooldown at less than 50°F/hour.
Continue RCS depressurization within the Natural Circulation Pressure-Temperature Limits curve.

Technical Reference OP 1102-11, Plant Cooldown, Rev. 122 page 29

Open Exam Reference

Learning Objective III.C.07.08 Given a set of RCS parameters/conditions determine if RCS head bubble formation has occurred and what actions are required to correct this occurrence.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .8, .10 ☐ 55.43 ☑ 55.45 .3

Discriminant Validity Statements

A Correct answer. Due to RV Head bubble formation the cooldown should be stopped and RCS pressure held constant or slightly higher.

B Plausible since stopping cooldown is correct, but depressurization could make the head bubble expand.

C Plausible due to maintaining the RCS pressure. The cooldown rate is the natural circulation limit, but the cooldown should be stopped.

D Plausible if the indication of RV Head bubble formation is missed, these are natural circulation cooldown limits.

Comments None.

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GPU NUCLEAR

TMI
Operating Procedure

Number: 1102-11

Title: **Plant Cooldown**

Revision No.: 122

Applicability/Scope: TMI Division

Responsible Office: Plant Operations Dir.

Effective Date: 12/17/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	D Jones	12-7-99
Procedure Owner	Thomas H. Gallup	12-14-99
PRG	[illegible]	12/16/99
Approver	[illegible]	12/14/99

ENCLOSURE 2

_____ 2.2 If a plant cooldown is required, establish a <50°F/hr cooldown rate and stay within the limits of the figure in this enclosure. These limits will prevent an RV head bubble formation.

> **NOTE**
>
> If void formation should occur in the RV head, the head bubble should be condensed before continuing the cooldown. If an RCP cannot be bumped, then <RCS pressure should be held constant or slightly higher until the head bubble has condensed as indicated by the return of pressure control to the pressurizer>.

_____ 2.3 Monitor the console indicators for RCS temperature, pressure and saturation margin to verify that at least a 25°F margin to saturation is being maintained. Increase makeup flow, increase cooling through the Steam Generator by decreasing OTSG pressure increasing level as necessary to maintain >25°F SCM and verify pressurizer heaters are available. Refer to EP 1202-29, "Pressurizer System Failure", if necessary to transfer pressurizer heaters to the backup power supply.

_____ 2.4 Verify natural circulation per the guidelines found in ATP 1220-10, "Abnormal Transients Rules, Guides and Graphs".

_____ 2.5 RCS forced flow cooldown is preferred over Natural Circulation Cooldown. If the Reactor Coolant Pump restart criteria is met, start one pump per loop per the guidelines in 1103-6, "Reactor Coolant Pump Operations", and then conduct a forced flow cooldown per the main body of this procedure.

> **NOTE**
>
> If pressurizer level increased due to formation of head bubble, do not reduce pressurizer level prior to starting pump. Starting a RCP may cause a decrease in Pressurizer level and subcooling margin due to possible head void collapse. Increase RCS make-up as necessary to maintain subcooling margin > 25°F and Pressurizer level ≥ 100".

> **NOTE**
>
> If Natural Circulation Cooldown is required, the steam generator emergency feedwater system (motor-driven pumps are preferable) should be used to control OTSG level in order to eliminate the RCS uncontrolled heat loss to the steam driven emergency feedwater pumps.

_____ 2.6 RCS pressure decrease, without spray flow, will be a result of pressurizer ambient losses (projected to be ~ 50 psig/hr without pressurizer heaters) and cooling of the pressurizer due to insurges of cooler reactor coolant. Reduce the RCS cooldown rate as necessary to prevent violation of the NDT limit.

_____ 2.7 Continue cooldown using natural circulation and the appropriate steps of Section 3.0 in the main body of this procedure. Adhere to the limits for RCS pressure and temperature during Natural Circulation as described in this enclosure.

Form ES-401-6

Written Exam Question Worksheet

Q # 004

Examination Outline Cross-Reference RO SRO

KA # APE 024 AK1.02 Page 4.2-14 Tier # 1

RO/SRO Importance Rating 3.6 3.9 Group # 1

Measurement Knowledge of the operational implications of the following concepts as they apply to Emergency Boration: Relationship between boron addition and reactor power

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** A

What is the goal of Emergency Boration following a reactor trip with a control rod stuck out?

Add Boric acid to the RCS until

A. 1% dk/k shutdown is achieved for the expected plant condition.

B. 1% dk/k shutdown is achieved assuming another rod stuck out.

C. 1% dk/k shutdown margin is achieved for the expected plant condition.

D. 1% dk/k shutdown margin is achieved assuming another rod stuck out.

Technical Reference ATOG 1210-10, Abnormal Transients Rules, Guides and Graphs, Rev. 37 page 6

Open Exam Reference

Learning Objective V.E.01.02 State and explain the Immediate Action(s) of ATP 1210-1, Reactor Trip, in the required order.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .8, .10 ☐ 55.43 ☑ 55.45 .3

Discriminant Validity Statements

A Correct answer.

B Plausible, defines minimum shutdown margin.

C Plausible with a misconception that 1% shutdown and 1% shutdown margin are equivalent.

D Plausible combination of the above.

Comments None.

FOR INFORMATION ONLY

GPU NUCLEAR	TMI Abnormal Transient Procedure	Number 4-4 1210-10

Title

Abnormal Transients Rules, Guides and Graphs

Revision No. **37**

Applicability/Scope	Responsible Office	Effective Date
TMI Division	Plant Operations Dir.	10/15/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	[illegible]	10-14-99
Procedure Owner	James N Paulles	10/15/99
PRG	[illegible]	10/15/99
Approver	James N Paulles	10/15/99

1.6.1 If 25°F subcooling margin is lost raise level in the operable OTSG(s) to 75-85% (Operating Range).

1.6.2 Maintain OTSG level ≥ 25" start-up range with RCPs on.

1.6.3 Maintain OTSG level ≥ 50% operate range with RCPs off.

1.7 Superheat Determination/Limit - Rule

1.7.1 Superheat exists at 25°F of superheat as determined by the value of Comp. Pt. C4132 (calculated Incore Superheat).

1.7.2 Superheat exists at 25°F of superheat as determined by the average of the five (5) highest, operable incore thermocouples (Comp. Pt. C4006) and RCS wide range pressure PI-949A (as read on panel PCL) or PT-963 (Comp. Pt.A0404).

1.7.3 Superheat exists at 25°F of superheat as determined by the highest operable BIRO incore thermocouple and RCS wide range pressure PI-949A (as read on panel PCL) or PT-963 (Comp. Pt. A0404).

2.0 GUIDELINES

2.1 Pressurizer Level Guide

If the reactor power is > 25% and pressurizer level cannot be maintained ≥ 150 inches, then trip the reactor and initiate HPI.

2.2 Emergency Boration Guide

Definition: Emergency Boration is the addition of boric acid to the RCS until 1% dk/k shutdown has been achieved for the expected plant condition as determined by Figure 1 of OP 1103-4, "Soluable Poison Concentration Control", or by OP 1103-15A, "Reactivity Balance".

2.2.1 Emergency Boration may be accomplished from any one of the following sources: (Refer to OP 1103-4 as necessary)

a. BWST

b. BAMT

c. RBAT

2.2.2 Emergency Borate using BWST

a. Open MU-V-14A or B

b. Establish a RCS letdown and bleed path if required to achieve the boration objective defined above.

Form ES-401-6

Written Exam Question Worksheet

Q # 005

Examination Outline Cross-Reference RO SRO

KA # APE 026 AA1.04 Page 4.2-18 Tier # 1

RO/SRO Importance Rating 2.7 2.8 Group # 1

Measurement Ability to operate and/or monitor the following as they apply to the Loss of Component Cooling Water: CRDM high-temperature alarm system

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

Which ONE (1) of the following AUTOMATIC actions should occur if Control Rod Drive cooling water outlet temperature exceeds 160°F?

A. MU-V-1A/B close

B. MU-V-3 closes

C. IC-V-1A/B close

D. Standby IC-P-1 starts

Technical Reference EP 1202-8, CRD Equipment Failure, Rev. 53 page 26

Open Exam Reference

Learning Objective V.D.03.01 State the Immediate Automatic Actions that will occur for the following CRD equipment failures/malfunctions per EP 1202-8:
d. Stator High Temperature

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .5, .6

Discriminant Validity Statements

A Correct answer.

B Plausible because this would serve the same purpose as the correct answer and MU-V-3 does get other high temperature closure signals.

C Plausible because there is an interlock with MU-V-1A/B.

D Plausible because this may improve cooling, these pumps have a different auto-start signal and are part of the manual actions.

Comments None.

GPU NUCLEAR	TMI Emergency Procedure	Number 1202-8
Title: **CRD Equipment Failure**		Revision No. 53
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Director	Effective Date: 10/12/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

TRAINING DEPT.

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	Signature	Date
Originator		10/7/99
Procedure Owner	Stuart Brantley	10-8-99
PRG		10/9/99
Approver	James N Bulls	10/8/99

E. **STATOR HIGH TEMPERATURE**

1E. Symptoms

1. Computer alarms and printouts of the affected mechanism(s) giving the stator temperature.

2. Possible low ICCW CRD cooling flow as indicated on IC10-FI or MAP alarm C-1-2.

3. Possible ICCW CRD filter dP High alarm on MAP C-1-4.

4. Possible ICCW cooler outlet temperature high on IC6-TI or MAP alarm C-2-3.

5. Possible CRD cooling water outlet temperature high on IC9-TI.

6. Possible IC system CRD cooling water outlet temperature high alarm on MAP C-1-3.

2E. Immediate Actions

1. Automatic Actions

a. MU-V-1A/B close when CRD cooling water outlet temperature reaches 160°F.

2. Manual Actions

NOTE
The steps with an asterisk (*) will be re-verified as the first step of the follow-up actions.

*a. Verify MU-V-1A/B close if CRD outlet temperature reaches 160°F.

*b. Check the ICCW system for proper operation by verifying

1. The following parameters for the CRD system:

i. Flow to CRD stators > 100 gpm.
ii. CRD return temperature < 160°F
iii. dP across IC-F-1A/B.

c. If required to provide continued cooling to the CRDMs, then start the standby IC-P-1 and/or valve in the standby IC-F-1.

3E. Follow-up Actions

Objective: The objective of this procedure is to prevent stator failure due to a loss of cooling water flow or high temperature in the ICCW system.

1. Reverify the steps in the Immediate Manual Actions that are marked with an asterisk (*). Use redundant indications where available.

Form ES-401-6

Written Exam Question Worksheet

Q # 006

Examination Outline Cross-Reference RO SRO

KA # APE 027 AA1.05 Page 4.2-20 Tier # 1

RO/SRO Importance Rating 3.3 3.2 Group # 1

Measurement Ability to operate and/or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: Transfer of heaters to backup power supply

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

Which ONE of the following outlines the basic steps for transferring a Pressurizer Heater Group from the normal power supply to an engineered safeguards bus?

A. Remove the disconnect device from the normal cabinet
Open the normal supply breaker
Install the disconnect device in the alternate cabinet
Rack in the engineered safeguards bus breaker

B. Open the normal supply breaker
Remove the disconnect device from the normal cabinet
Install the disconnect device in the alternate cabinet
Rack in the engineered safeguards bus breaker

C. Remove the disconnect device from the normal cabinet
Install the disconnect device in the alternate cabinet
Open the normal supply breaker
Rack in the engineered safeguards bus breaker

D. Open the normal supply breaker
Remove the disconnect device from the normal cabinet
Rack in the engineered safeguards bus breaker
Install the disconnect device in the alternate cabinet

Technical Reference EP 1202-29, Pressurizer System Failure, Rev. 54 page 20

Open Exam Reference

Learning Objective 010C01020 Swap pressurizer heater groups 8 or 9 to an ES bus.
1

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .5, .6

Discriminant Validity Statements

A Plausible due to correct steps, but wrong sequence.

B Correct answer.

C Plausible due to correct steps, but wrong sequence.

D Plausible due to correct steps, but wrong sequence.

Comments None.

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GPU NUCLEAR

TMI
Emergency Procedure

Number: 1202-29

Title: **Pressurizer System Failure**

Revision No.: 54

Applicability/Scope: TMI Division

Responsible Office: Plant Operations Dir.

Effective Date: 07/06/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

SIMULATOR ROOM
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	Signature	Date
Originator		6-21-99
Procedure Owner		6-24-99
PRG		6/29/99
Approver	James Naulls	6/25/99

EXHIBIT 5

Pressurizer Heater Transfer

A. To transfer Group 8 (Group 9) pressurizer heater power supply from 1B pressure heater control center to 1P (1S) engineered safeguard bus, the following sequence must be followed:

_____ 1. All keys are held in lock during normal operation. Verify that the transfer disconnect devices are in their normal position.

_____ 2. Manually trip the pressurizer heater control center 1B circuit breaker 2B (3A). This will permit removal of the kirk keys.

_____ 3. Remove keys K1 and L1 (K2 and L4), this prevents reclosing circuit breaker 2B (3A).

_____ 4. Take keys K1 and L1 (K2 and L4) to disconnect device S1 (S2).

_____ 5. Using key K1 (K2), open S1 (S2) enclosure door.

_____ 6. Turn key L1 (L4) to permit removal of disconnect assembly S1 (S2) and key L3 (L6).

_____ 7. Unbolt and remove disconnect assembly S1 (S2) and key L3 (L6). L1 (L4) remains inlock.

_____ 8. Lock enclosure door with key K1 (K2).

_____ 9. Take disconnect assembly S1 (S2), keys K1 and L3 (K2 and L6) to disconnect device S1 - alternate (S2 alternate).

_____ 10. With key K1 (K2) open S1 - alternate (S2 alternate) enclosure door.

_____ 11. Insert and bolt disconnect assembly S1 (S2) in place, turn L3 (L6), remove key L2 (L5).

_____ 12. Lock enclosure door with key K1 (K2), L3 (L6) remains in lock.

_____ 13. Take keys L2 (L5) and K1 (K2) to 1P (1S) 480V ES switchgear unit 3A.

_____ 14. Turn keys to permit switchgear unit 3A to be racked in. Keys L2 (L5) and K1 (K2) remain in lock.

_____ a. Rack in Unit 3A on the 1P (1S) 480V Switchgear.

Form ES-401-6

Written Exam Question Worksheet

Q # 007

Examination Outline Cross-Reference RO SRO

KA # BW E05 EK3.3 Page 4.3-10 Tier # 1

RO/SRO Importance Rating 4.2 3.8 Group # 1

Measurement Knowledge of the reasons for the following responses as they apply to the Excessive Heat Transfer: Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

The Control Room Supervisor directs stabilizing RCS pressure and T-cold on the "B" OTSG following a steam rupture on the "A" OTSG.

These actions are taken to prevent RCS heatup and repressurization in order to:

A. prevent possible loss of subcooling margin due to insurge and cooling of the Pressurizer.

B. minimize the formation of a reactor vessel head bubble due to the rapid cooldown

C. prevent placing additional stresses on the Reactor Coolant System.

D. minimize additional compressive stresses to the "A" OTSG tubes.

Technical Reference TDR No. 1199, Comparison of ATP 1210-3 and BWOG GEOG, Rev. 2 page 21

Open Exam Reference Steam Tables

Learning Objective V.E.03.02 Explain why it is desirable to minimize heatup and repressurization following an excessive primary to secondary heat transfer event.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5, .10 ☐ 55.43 ☑ 55.45 .6, .13

Discriminant Validity Statements

A Plausible effect on pressurizer, but not the basis.

B Plausible for natural circulation.

C Correct answer.

D Plausible effect on OTSG tubes, but not the basis.

Comments None.

GPU NUCLEAR	TDR No. 1199	Revision No. 2
Technical Data Report	Budget Activity No. 115403	Page 1 of 42
Project: TMI-1	Department/Section: Nuclear Analysis & Fuels/Safety & Risk Analysis	
	Revision Date 11/15/99	

Document Title: Comparison of ATP 1210-3 and BWOG GEOG

Originator Signature	Date	Approval(s) Signature	Date
		Approval for External Distribution	Date

Does this TDR include recommendation(s)? [ ] Yes [x] No If yes, TFWR/TR # _____

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Abstract:
This report provides a comparison of the TMI-1 Abnormal Transient Procedure ATP 1210-3, "Excessive Primary to Secondary Heat Transfer", with the B&W Owner's Group Generic Emergency Operating Guidelines. Differences are identified and justification provided for each difference.

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* Abstract Only

TMI

2.6 Minimize RCS heatup and repressurization. Evaluate for PTS, refer to ATP 1210-10, Guideline 2.13.

GEOG

[GEOG Step 1.0 Caution]

CAUTION

Excessive cooldown rate can invoke PTS guidance (Specific Rule 3.0). Once invoked, the PTS guidance remains in effect.

It should be noted that as long as SCM is adequate the core will be cooled. For this reason it is not necessary to continue taking actions to recover pressurizer level if SCM could become excessively large (this is true even if PTS guidance has not been invoked). Actions to initiate HPI to accommodate RC density increases (RC shrinkage) are appropriate, however, the primary parameter of interest is SCM not pressurizer level. That is, if SCM is adequate there is no need to attempt to regain pressurizer level during an overcooling. This can lead to excessive SCM and subsequent inventory/pressure and/or PTS problems.

Maintaining adequate but not excessive SCM will provide margin to overpressure limits should conditions change.

Discussion:

Priority is given to stopping the overcooling and then to stabilizing the RCS and addressing PTS. Therefore, PTS guidance is placed later in the ATP than in the GEOG. HPI control while SCM exists is rule-based in accordance with ATP 1210-10, "High Pressure Injection (HPI) Throttling Criteria," which refers to the PTS guidance. PTS is implemented as a Guide in ATP 1210-10.

OS-24 states:

"4.1.4 Rules: Rules comprise the first section of procedure steps in ATP 1210-10, Abnormal Transients Rules, Guides and Graphs. In general, they are actions directly related to core cooling. The Rules must be performed regardless of time of occurrence or location in the procedures. Therefore, the Rules must be memorized by the operators. The development of specific actions required to be performed whenever the conditions are met allows for simplification of the procedures. This allows quick performance because they are governed by rule-based behavior. Rule-based behavior assures the correct actions are taken in a timely manner."

Form ES-401-6

Written Exam Question Worksheet

Q # 008

Examination Outline Cross-Reference RO SRO

KA # APE 051 AA2.02 Page 4.2-34 Tier # 1

RO/SRO Importance Rating 3.9 4.1 Group # 1

Measurement Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

Current plant conditions:
- Reactor is operating at 100% power.
- 6 CW Pumps are operating.
- Winter conditions exist.

Which ONE (1) statement describes when the CRO is required to manually trip the turbine? Assume no automatic reactor trip.

A. Loss of one CW Pump.

B. Condenser pressure is 8.7 inches Hg absolute.

C. Loss of the Gland Steam Exhauster.

D. Condenser pressure is 7.7 inches Hg absolute.

Technical Reference OPM, Main Turbine, H-01, Page 46, Rev 13

Open Exam Reference

Learning Objective IV.D.04.12 Describe operator actions in response to decreasing main or auxiliary condenser vacuum.

Question Source ☐ New ☑ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☑ Used in Last Two TMI NRC Exams Exam Date 1998 SRO EXAM

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5 ☑ 55.45 .13

Discriminant Validity Statements

A Plausible distracter considering summer time operations. Question stem establishes winter conditions, when we often run with 5 pumps.

B Correct answer. Automatic setpoint is 8.5 inches Hg Absolute if >75% power, 7.5 inches Hg Absolute if <75% power.

C Plausible distracter due to connections with Turbine Gland Seal system.

D Plausible misconception that setpoint is 7.5 inches rather than 8.5 inches Hg Absolute (power level dependent).

Comments None.

FOR INFORMATION ONLY

SECTION H-01
REVISION 13

Q-08

TMI OPERATIONS PLANT MANUAL
MAIN TURBINE

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Concur

Owner — Charles C. Seitz — Date 12-2-99

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Training Coordinator, TMI — Linda L. Bitter — Date 12-14-99

Approved

Plant Operations Director TMI — D W [illegible] — Date 12-15-99

4.3.2 Main Turbine Generator Trips

a. Generator Fault

b. Reactor Trip

c. Both Feed Pumps Tripped

d. Moisture Separator Level Hi-Hi (after 12 second delay) (N-1-1)

e. Main Condenser Vacuum High (2 of 3)

f. Loss of Both Main Turbine Speed Signals (2 of 3)

g. Turbine Overspeed

h. Turbine Backup Overspeed

i. EHC Hydraulic Oil Pressure Low (2 of 3)

j. Turbine Thrust Bearing Failure (2 of 3)

k. Stator Coolant High Outlet Temperature (2 of 3)

l. Stator Coolant Low Inlet Pressure (2 of 3)

m. Stator Coolant Low Flow (2 of 3)

4.3.3 Generator Fault - Any fault condition such as load loss, short circuits, under or over excitation which will trip the generator and or field breakers open will cause a turbine trip.

4.3.4 Reactor Trip - Anytime the reactor is tripped a signal is sent to the turbine trip system which will trip the turbine.

4.3.5 Main Feedwater Pump Trip - If both feedwater pumps are operating and for some reason they trip, a signal is sent to the turbine trip system which will trip the turbine.

4.3.6 Moisture Separator High Level - Any water level in MO-T-2A-F. There is a 12 second delay between receiving alarm N-1-1 and the turbine tripping.

4.3.7 Main Condenser Vacuum High – If main condenser backpressure raises to 7.5" HgA for loads ≤75%, or 8.5" HgA for loads > 75%, a trip signal is initiated to trip the turbine.

4.3.8 Loss of Main Turbine Speed Signals - See Electrohydraulic Control System

4.3.9 Turbine Overspeed - The mechanical overspeed trip device in the front standard is the first line of defense against overspeed protection. If the turbine speed reaches greater than 108% of rated speed but less than or equal to 112 percent of rated turbine speed a mechanical signal (movement of linkage) is sent to the mechanical trip valve in the emergency trip system which will trip the turbine. The mechanical signal is originated from the mechanical overspeed trip device. See the electrohydraulic control system handout for more details.

Form ES-401-6

Written Exam Question Worksheet

Q # 009

Examination Outline Cross-Reference RO SRO

KA # EPE 055 GEN EP 2.4.31 Page 2-14 Tier # 1

RO/SRO Importance Rating 3.3 3.4 Group # 1

Measurement Station Blackout Knowledge of annunciators alarms and indications, and use of the response instructions

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** A

The following plant conditions exist:
- Substation Panel (SS-1) 230 KV bus meters indicate zero volts
- Main Generator Output Breakers are open

Which ONE (1) of the following is a required operator action?

A. Restore normal makeup and seal injection.

B. Control MS-V-4A/B from the backup loaders.

C. Start the Generator DC emergency seal oil pump.

D. Provide the EF-V-30s a Local Auto setpoint of 50%.

Technical Reference EP 1202-2, Loss of Station Power, Rev. 49 page 3
ATOG 1210-1, Reactor Trip, Rev. 40 page 3

Open Exam Reference

Learning Objective V.D.01.02/ V.E.01.08 Given a list of symptoms, identify the event, determine what Immediate Automatic Actions will occur and what Immediate Manual actions are required IAW EP 1202-2.
Given the plant conditions determine which IF/THEN step(s) of ATP 1210-1 apply and which other procedure(s) will be used in conjunction with the ATP 1210-1.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .10 ☐ 55.43 ☑ 55.45 .3

Discriminant Validity Statements

A Correct answer.

B Plausible, this is required under different circumstances.

C Plausible because this pump is required, but it starts automatically.

D Plausible, the operator may desire manual operation for throttling purposes, but the setpoint occurs automatically.

Comments None.

FOR INFORMATION ONLY

GPU NUCLEAR	TMI Emergency Procedure	Number 1202-2
Title: **Loss of Station Power**		Revision No. 49
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: 08/20/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	Henry B Shipman	8/11/99
Procedure Owner	Ronald S. Campbell	8-16-99
PRG	[illegible]	8/17/99
Approver	[illegible]	8/17/99

1.0 **SYMPTOMS**

A. All AC lighting in the Control Room goes out. (DC lights come on.) AC lights come back on in approximately 10 sec if Emergency Diesels load.

B. Substation Panel (SS-1) 230 KV bus meters indicates zero (0) volts.

C. Main Generator Output Breakers open. (Amber-Green lights)

D. Potential 13.2 KV loss PDMS alarms

2.0 **IMMEDIATE ACTION**

A. Automatic Actions

1. Reactor/Turbine trips.
2. Control Room DC lighting comes on.
3. Generator DC emergency seal oil pump starts.
4. DC lube oil pumps start for Main and Feedwater pump turbines.
5. EFW initiates to both OTSG's with control set at 50% on the operating range.
6. Screenhouse and Circ water pump house diesel fire pumps start. (FS-P-3 will not auto start if ISPH power is restored in < 30 seconds.)
7. Substation Diesels (EG-Y-3A/B) start and power PM-1 and PM-2.
8. D & E 4160V busses are energized in 10 secs. if EG-Y-1A/B start.

B. Manual Actions

1. Perform the Immediate Manual Actions of 1210-1. Reactor Trip (DSS breakers will not open until off-site power is restored.)

3.0 **FOLLOW-UP ACTION**

Objective:

Ensure the core is in a stable condition by establishing Natural Circulation, maintaining adequate Shutdown Margin, and minimizing inventory loss and RCS shrink. This procedure also provides a method for restoring power and plant equipment to service when available.

_____ 3.1 Refer to Attachment 9, and if any Emergency Action Levels have been exceeded, implement EPIP-TMI-.01, EMERGENCY CLASSIFICATIONS, and additional procedures as required.

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GPU NUCLEAR	TMI Abnormal Transient Procedure	Number 1210-1
Title **Reactor Trip**		Revision No. 40
Applicability/Scope TMI Division	Responsible Office Plant Operations Dir.	Effective Date 10/19/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

SIMULATOR ROOM FILE COPY

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	Signature	Date
Originator	[illegible]	10/15/99
Procedure Owner	James N Paulls	10/15/99
PRG	John S. Schaub	10/15/99
Approver	James N Paulls	10/15/99

2.3 Control Main Feedwater.

Verify FW-V-5A/B stroking closed or are closed; AND ICS controlling MFW flow.

IF ICS is not controlling MFW flow;	THEN take Hand control of the FW regulating valves and/or MFW pumps AND control proper OTSG level and valve DP.
IF MFW flow is still excessive;	THEN trip both MFW pumps, **PRIOR TO** exceeding 97.5% OTSG level.

2.4 Verify ICS/NNI Power On (PCL).

IF both AUTO AND HAND power lights are off;	THEN trip MFW pumps, AND verify EFW, AND maintain primary to secondary heat transfer by using the ADV B/U loaders to control OTSG pressure. AND Refer to EP 1202-40.
IF any other lights or combination of lights are off;	THEN refer to EP 1202-40/41/42.

2.5 Verify 4160 volt buses 1A, 1B, 1C, 1D and 1E energized (CR and PR).

IF loss of station power has occurred;	THEN verify starting and loading of at least one D/G. AND Restore Make-up, seal injection and EFW. AND Refer to EP 1202-2.
IF both ES buses are deenergized;	THEN isolate letdown AND seal return, AND verify steam driven EFW pump operating AND Refer to EP 1202-2.

2.6 Start 2nd Make-up Pump and open MU-V-217.

IF MU Tank is less than 55";	THEN open MU-V-14A or B as necessary to maintain MU Tank level greater than 55".
IF unable to maintain pressurizer level greater than 20 inches;	THEN initiate HPI.

Form ES-401-6

Written Exam Question Worksheet

Q # 010

Examination Outline Cross-Reference RO SRO

KA # APE 062 AA2.02 Page 4.2-29 Tier # 1

RO/SRO Importance Rating 2.9 3.6 Group # 1

Measurement Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The cause of possible SWS loss

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

Based on expected plant response, identify the ONE (1) event below that would result in AUTOMATIC isolation of the Nuclear Service Closed Cooling Water to the Reactor Building.

A. Large Break LOCA at the discharge of RC-P-1D.

B. Inadvertent train "B" 4 psig ESAS actuation signal.

C. Loss of Nuclear Service River Water cooling system.

D. Pipe rupture on the NS supply header to the RCP coolers

Technical Reference OPM Sec B-11page 9, Rev.7.

Open Exam Reference

Learning Objective IV.B.05.11 Identify/describe the effects of a Line Break Isolation signal on the NSCCW system including the purpose, actuation conditions, components affected and the two methods of resetting the actuation.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge

☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☐ 55.43 ☑ 55.45 .13

Discriminant Validity Statements

A Correct answer.

B Plausible since this provides one half of the line break isolation signal.

C Plausible since this provides cooling for NSCC system.

D Plausible since this provides one half of the line break isolation signal.

Comments None.

FOR INFORMATION ONLY

SECTION B-11
REVISION 8

Q-010

TMI OPERATIONS PLANT MANUAL
NUCLEAR SERVICES CLOSED COOLING WATER

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Concur

Owner R. E. Campbell Date 11-7-97

Operations Training Manager, TMI R H Carr Date 11/7/97

Training Coordinator, TMI Lindsay L. Kettle Date 11-17-97

Approved

Plant Operations Director TMI L. G. Nole Date 11-20-97

8.1 Line Break Isolation

Pushbuttons are provided on console center and console right for resetting the line break isolation circuit. Console center pushbutton is for Actuation 'A' which closes NS-V-4 and NS-V-15 while the console right pushbutton is for Actuation 'B' which closes NS-V-15 and NS-V-35.

During normal operation when a low surge tank level is received the amber half of the pushbutton will be lit. When the tank level is returned to normal the reset pushbutton is depressed and the amber light goes out.

When ESAS actuates (1600 psig) and a low surge tank level exist this will de-energize the amber light and the blue actuation light will energize, and the valves will close (NS-V-4, 15, 35). In order to reopen the valves either the surge tank level must be returned to normal and the reset button depressed or the HPI. signal must be bypassed.

Basically if only the amber half of the button is lit this indicates a reset permit and if the blue half is lit this indicates actuation.

8.2 DW-P-1

A start-stop control switch is located on console center next to the surge tank level indicators which allows starting of DW-P-1. DW-P-1 must be used to fill the surge tank due to the surge tank pressure and the surge tank elevation being too great a head to be overcome by DW-P-2.

Emergency makeup can also come from the condensate system.

9.0 INTERLOCKS

9.1 Standby pump automatically starts on loss of the running pump.

9.2 Valves NS-V-4, 15, 35 shut on 30 psig ES signal or on line break isolation.

9.3 The selected nuclear services pump will start on ES signal (BLOCK 3).

9.4 NS-V-16A/B automatically open when their respective Spent Fuel Cooling pump is started.

9.5 NS-V-52A, B, C and NS-V-53A, B, C will open automatically as its respective RB fan motor AH-E-1A,B,C operates.

9.6 Operation of 43 selector switches.

9.6.1 NS-P-1B can be powered from either P or S 480 V bus.

9.6.2 A selector switch is used to select which pumps starts on ES actuation.

9.6.3 On 1P bus the selector switch can be positioned to the 1A or 1B position. If the switch is left in the 1B position the 1B pump will start on ES.

9.6.4 The same is true for the 1S bus where either the 1B or 1C pumps can be selected. If the 1B pump is selected on either bus the switch handle is captured.

Form ES-401-6

Written Exam Question Worksheet

Q # 011

Examination Outline Cross-Reference RO SRO

KA # APE 067 AK3.04 Page 4.2-52 Tier # 1

RO/SRO Importance Rating 3.3 4.1 Group # 1

Measurement Knowledge of the reasons for the following responses as they apply to the Plant Fire on Site: Actions contained in EOP for plant fire on site

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** C

Current plant conditions:
- Numerous fire alarms are actuated on panels HVB, PLA, and PLB.
- Fire dampers AH-D-4 and AH-D-5 have automatically closed.
- Air tunnel deluge systems have actuated.
- Air tunnel halon system has actuated.
- Aux and Fuel Handling Building ventilation supply fans have tripped.

Which ONE (1) statement describes the action(s) required for this condition?

A. Trip Aux and Fuel Handling Building ventilation exhaust fans.

B. Actuate Relay Room CO2 system manually.

C. Start three fire pumps.

D. Trip the reactor.

Technical Reference EP 1202-31, Fire Rev 58 page 5.

Open Exam Reference

Learning Objective V.D.13.03 State the Immediate Manual Actions required for a fire per EP 1202-31.

Question Source ☐ New ☑ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☑ Used in Last Two TMI NRC Exams Exam Date 1998 SRO EXAM
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5, .10 ☐ 55.43 ☑ 55.45 .6, .13

Discriminant Validity Statements

A Plausible distracter since this would terminate ventilation effects on the fire.

B Plausible distracter since this is a correct action for another fire scenario.

C Correct answer.

D Plausible distracter since this is a correct action for a relay room fire scenario, and since Air Intake Tunnel dampers have actuated..

Comments None.

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AmerGen	TMI - Unit 1 Emergency Procedure	Number 1202-31
Title **Fire**		Revision No **58**
Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date 04/14/00

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

TRAINING DEPT.

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	Signature	Date
Originator	RE Seaglitz	4/13/00
Procedure Owner	FEE RSCampbell	4/13/00
PRG		4/14/00
Approver		4/13/00

D. Verify automatic starting of fire pumps, if required, by observing illuminated red RUNNING indicator light(s) on Panel PL for FS-P2, 3, and FS-P-1

NOTE

1. Fire pumps will also start on low fire main pressure as indicated below:

Motor Driven Fire Pump	(FS-P-2)	(90 psig)
Diesel Driven (River) Fire Pump	(FS-P-3)	(80 psig)
Diesel Driven (Flume) Fire Pump	(FS-P-1)	(70 psig)

2. The announcement of any fire in an enclosed "Out Building" or any other area requiring admittance by a security guard shall require the roving security officer to report to the scene and provide the necessary entry.

E. Start all non-operating fire pump(s) if required on Panel PL and verify that pump(s) start.

F. Make a repeat announcement of the following using the page channel and over the Ops. radio system:

"Fire, (location and unit), (type), Fire Brigade Report to (location)

G. Turn off the whelen siren.

NOTE

Any fire in TMI-2 (inside the TMI-1 Protected Area) which cannot be readily controlled by portable fire extinguishing equipment will require off-site assistance (due to the PDMS status of the water systems). If needed, the call for off-site assistance should be made as soon as possible.

H. Refer to OP 1104-45P, "Fire Mitigation" for Equipment Operational Concerns by Fire Zone/Area and other actions which may be necessary to mitigate the consequences of fire.

I. For TMI-2, refer to OP 1104-45Q, "TMI-2 Fire Protection System" regarding operation of TMI-2 fire protection equipment. The TMI-2 Protectowire fire detection alarm responses are covered in PC-F, "PC Fire Panel Alarm Response". OP 1104-45P provides guidance on specific responses such as for the TMI-2 Reactor Building.

Form ES-401-6

Written Exam Question Worksheet

Q # 012

Examination Outline Cross-Reference RO SRO

KA # APE 068 AK2.03 Page 4.2-54 Tier # 1

RO/SRO Importance Rating 2.9 3.1 Group # 1

Measurement Knowledge of the interrelations between the Control Room Evacuation and the following: Controllers and positioners

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

The Control Room has been evacuated and you are stationed at the Remote Shutdown Panels.

Which ONE of the following sets of controls are available at the Remote Shutdown Panels to control OTSG level and pressure?

A. EF-V-30A and B only
MS-V4A and B

B. EF-V-30C and D only
MS-V-3A through F

C. EF-V-30A, B, C and D
MS-V-4A and B

D. EF-V-30A, B, C and D
MS-V-3A through F

Technical Reference EP 1202-37, Cooldown from Outside the Control Room, Rev. 52 pages 4 and 5

Open Exam Reference

Learning Objective IV.E.18.16 Given a plant parameter, explain how it is monitored and controlled during a remote shutdown sequence.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .7

Discriminant Validity Statements

A Plausible because EF-V-30A and B will feed both OTSGs, but all of the EF-V-30s are available.

B Plausible because EF-V-30C and D will feed both OTSGs, but all of the EF-V-30s are available. The MS-V-3s will steam the OTSGs, but are not available at the RSD Panels.

C Correct answer. All of the EF-V-30s are available for feeding and the MS-V-4s for steaming the OTSGs.

D Plausible because all of the EF-V-30s are available for feeding. The MS-V-3s will steam the OTSGs, but are not available at the RSD Panels.

Comments None.

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GPU Nuclear	TMI-1 Emergency Procedure	Number 1202-37
Title: **Cooldown from Outside the Control Room**		Revision No. 52
Applicability/Scope: TMI Division	Responsible Office: Plant Ops. Dir.	Effective Date: 12/15/98

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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Originator	[illegible]	12/9/98
Procedure Owner	L. G. Noll for R. S. Campbell	12-10-98
PRG	[illegible]	12-10-98
Approver	L. G. Noll	12-10-98

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_____ c. "DH CLSD CYC CLG Pump DC-P-1B" - isolates DC-P-1B from Control/Relay Room ckts.

_____ d. "Reactor Coolant" - isolates RC-V-2 and RC-V-3 from Control/Relay Room ckts.

_____ e. "Makeup and Purif." - isolates MU-V-2A/B, MU-V-18, MU-V-20, MU-P-1C, MU-V-16C/D, MU-V-37, MU-V-14B, MU-P-3B, MU-P-3C, MU-V-8, and MU-V-32 Flow Status Lights from Control/Relay Room ckts.

_____ f. "Nuc. Serv. River Wtr" - isolates NR-V-1C, NR-V-15B, NR-V-18 and NR-P-1C from Control/Relay Room ckts.

_____ g. "DH River Water" - isolates DR-V-1B and DR-P-1B from Control/Relay Room ckts.

_____ h. "Comm System" - isolates Gray Page, M & I and Substation Merge Isolate Switch from Control/Relay Room ckts.

_____ i. "Intermediate Clg" - isolates IC-V-2, IC-V-4, and IC-P-1B from Control/Relay Room ckts.

_____ j. "Nuc. Serv. Clsd. Cyc. Clg." - isolates NS-P-1C from Control/Relay Room ckts.

_____ k. "RB Emerg. Clg. Riverwater" - isolates RR-V-1B from Control/Relay Room ckts.

_____ l. "Feeder to 1T Swgr." - isolates 1T-02 from Control/Relay Room ckts.

_____ m. "Feeder to 1B Screenhouse MCC" - isolates 1T-4C from Control/Relay Room ckts.

_____3.5 At 1BES MCC, Unit 5B. position IA-P-1B transfer switch to "emergency", and start IA-P-1B with local start switch.

_____3.6 Use Key No. 98 to open RSTSP "C" and switch the one switch to "emergency".

_____ a. "Makeup System" - isolates MU-P-1B from Control/Relay Room ckts.

NOTE

It is important to gain OTSG pressure and level control as soon as possible. If EF-P-2B is not operating at this time continue to closely monitor OTSG pressure and level until EF-P-2B is made available.

NOTE

Use this procedure in conjunction with Plant Cooldown (1102-11) and Plant Shutdown (1102-10) to aid in gaining control of secondary plant systems.

____3.7 At RSD panel A, establish control of EF-V-30A (A OTSG) and EF-V-30C (B OTSG) and MS-V-4A (A OTSG) to stabilize plant and establish/control primary to secondary heat transfer.

_____3.8 At RSD panel B, establish control of EF-V-30B (B OTSG) and EF-V-30D (A OTSG) and MS-V-4B (B OTSG) to stabilize plant and establish/control primary to secondary heat transfer.

_____3.9 At the 1C ES VLV emergency isolation transfer switch cabinet, use CAT 60 key (Key No. 21) to open the door and place the transfer switch to "emergency".

_____3.10 At the 1C ES VLV normal transfer switch cabinet, ensure it is selected to the 1S Bus (Source 2) by going to test on the 1P: Bus (Source 1) toggle switch.

_____3.11 Verify the status of the 1E 4160V Bus. (White light on RSD "B" Panel)

_____ 3.11.1 If the 1E 4160V bus is already energized (white light on RSD "B" Panel) do not start the 1B D/G. Page the operator at the 1E Bus to perform the following:

a. Using 69 key, go to emergency on the following components

_____ 1. 1SB-E2

_____ 2. EF-P-2B

_____ 3. S1-02

_____ 4. DH-P-1B

_____ 5. RR-P-1B

_____ 6. T1-02

_____ 7. 1SA-E2

NOTE

If off-site power is lost subsequently, the diesel must be started per Step 3.11.2.

_____ b. Go to Step 3.12.

_____ 3.11.2 If the 1E 4160V bus is not energized (no white light at RSD "B" Panel), then page the operator at the 1E bus to perform the following:

a. Using 69 key, go to emergency on the following components

_____ 1. 1SB-E2

_____ 2. EF-P-2B

_____ 3. S1-02

_____ 4. DH-P-1B

Form ES-401-6

Written Exam Question Worksheet

Q # 013

Examination Outline Cross-Reference RO SRO

KA # APE 069 AA1.01 Page 4.2-57 Tier # 1

RO/SRO Importance Rating 3.5 3.7 Group # 1

Measurement Ability to operate and/or monitor the following as they apply to the Loss of Containment Integrity: Isolation valves, dampers, and electropneumatic devices

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

Which ONE (1) of the following would constitute a loss of containment integrity during refueling operations?

A. Reactor Building purge in progress.

B. An open penetration is sealed with fire foam.

C. One airlock door on the personnel hatch open.

D. The equipment hatch is secured with four (4) bolts.

Technical Reference Tech. Specs. page 3-44

Open Exam Reference

Learning Objective V.B.01.05 State the Containment Integrity requirements for moving fuel during a refueling outage.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .5, .6

Discriminant Validity Statements

A Plausible, the examinee may consider purging not allowed while refueling.

B Correct answer. This is a low pressure containment closure requirement.

C Plausible, the examinee may consider normal containment integrity requirements.

D Plausible, the examinee may consider four bolts insufficient.

Comments None.

3.8 FUEL LOADING AND REFUELING

Applicability : Applies to fuel loading and refueling operations.

Objective : To assure that fuel loading and refueling operations are performed in a responsible manner.

Specification

3.8.1 Radiation levels in the Reactor Building refueling area shall be monitored by RM-G6 and RM-G7. Radiation levels in the spent fuel storage area shall be monitored by RM-G9. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service.

3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

3.8.3 At least one decay heat removal pump and cooler shall be operable.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required for refueling shutdown.

3.8.5 Direct communications between the control room and the refueling personnel in the Reactor Building shall exist whenever changes in core geometry are taking place.

3.8.6 During the handling of irradiated fuel in the Reactor Building at least one door in the personnel and emergency air locks shall be closed. The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

3.8.7 During the handling of irradiated fuel in the Reactor Building, each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange or manual valve, or

2. Be capable of being closed by an operable automatic containment purge and exhaust isolation valve.

3.8.8 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

Form ES-401-6

Written Exam Question Worksheet

Q # 014

Examination Outline Cross-Reference RO SRO

KA # EPE 074 EA2.04 Page 4.1-17 Tier # 1

RO/SRO Importance Rating 3.7 4.2 Group # 1

Measurement Ability to determine or interpret the following as they apply to a Inadequate Core Cooling: Relationship between RCS temperature and main steam pressure

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

A reactor trip has occurred and the crew is verifying the immediate actions of ATP 1210-1, Reactor Trip.

The following plant conditions exist five (5) minutes post-trip:

- RCS Pressure 1900 psig

	Loop "A"	Loop "B"
- RCS Thot	590°F	591°F
- RCS Tcold	587°F	586°F
- OTSG Pressure	1010 psig	750 psig
- OTSG Level	11"	12"

- RM-A-5 4.79E4 cps
- RM-A-15 3.62E4 cps

Which ONE of the following ATOG procedures should the crew GO TO?

A. ATP 1210-2, Loss of 25°F Subcooled Margin

B. ATP 1210-3, Excessive Primary to Secondary Heat Transfer

C. ATP 1210-4, Lack of Primary to Secondary Heat Transfer

D. ATP 1210-5, OTSG Tube Leakage

Technical Reference ATOG 1210-1, Reactor Trip, Rev. 40 page 4

Open Exam Reference Steam Tables

Learning Objective V.E.01.08 Given the plant conditions determine which IF/THEN step(s) of ATP 1210-1 apply and which other procedure(s) will be used in conjunction with the ATP 1210-1.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge

☑ Comprehension or Analysis

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5 ☑ 55.45 .13

Discriminant Validity Statements

A Plausible due to elevated RCS temperatures.

B Plausible due to reduced RCS pressure, and reduced "B" OTSG pressure.

C Correct answer.

D Plausible due to indication of OTSG tube leakage, but lower priority than correct answer.

Comments None.

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GPU NUCLEAR	TMI Abnormal Transient Procedure	Number 1210-1
Title **Reactor Trip**		Revision No. **40**
Applicability/Scope TMI Division	Responsible Office Plant Operations Dir.	Effective Date 10/19/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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Originator	[illegible]	10/15/99
Procedure Owner	James N. Faulls	10/15/99
PRG	John S. Schork	10/15/99
Approver	James N. Faulls	10/15/99

2.7 Verify Safety System Status.

Verify RCS greater than 1600 psig and RB less than 4 psig.

IF an ES actuation setpoint has been reached;	THEN verify all components have been actuated for the respective ES signal(s).

Verify OTSG greater than 600 psig and 10 inches level.

IF an HSPS actuation setpoint has been reached;	THEN verify all components have been actuated for the respective HSPS signal(s).

2.8 Announce Reactor Trip over plant page and radio.

2.9 Verify SCM ≥ to 25°F.

IF SCM < 25°F	THEN trip all RCP's, AND initiate HPI, AND verify EFW, AND raise OTSG level to 75-85%, AND go to ATP 1210-2.

2.10 Verify RCS temperature/pressure and OTSG pressure approaching post trip temperatures and pressures within 2-5 minutes.

IF excessive primary to secondary Heat Transfer	THEN throttle MFW/EFW, AND isolate steam leak, AND increase RCS make-up as necessary AND in the absence of higher priority symptoms go to ATP 1210-3.
IF lack of primary to secondary Heat Transfer	THEN verify MFW/EFW AND in the absence of higher priority symptoms go to ATP 1210-4.

2.11 Verify indications of OTSG tube leakage do NOT exist.

IF an OTSG Tube Leak/ Rupture has occurred;	THEN in the absence of higher priority symptoms go to ATP 1210-5.

Form ES-401-6

Written Exam Question Worksheet

Q # 015

Examination Outline Cross-Reference RO SRO

KA # APE 076 AK3.06 Page 4.2-58 Tier # 1

RO/SRO Importance Rating 3.2 3.8 Group # 1

Measurement Knowledge of the reasons for the following responses as they apply to the High Reactor Coolant Activity: Actions contained in EOP for high reactor coolant activity

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

The plant announcement in the IMA section of 1202-11, "High Activity in Reactor Coolant", ensures that:

A. personnel man required stations for plant shutdown.

B. personnel leave the controlled area to minimize exposure.

C. chemistry bypasses the RM-G-18 interlock for CA-V-1,2,3 and 13.

D. the primary AO takes action to open MU-V-2A/B to restore inventory control.

Technical Reference EP 1202-11, High Activity in Reactor Coolant, page 2 Rev 25.

Open Exam Reference

Learning Objective None None

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5, .10 ☐ 55.43 ☑ 55.45 .6, .13

Discriminant Validity Statements

A Plausible since plant shutdown is required if letdown cannot be restored.

B Correct answer.

C Plausible since this is done to allow sampling of the RCS.

D Plausible if the misconception exists that letdown must be restored.

Comments There is no specific TMI objective associated with knowing the basis for the immediate actions of this procedure. The closest objective, V.D.06.02 with a rating of 3.6, reads "State the immediate manual actions required for High Activity in the Reactor Coolant System IAW EP 1202-11." Since the announcement itself is self explanatory, AMERGEN feels this question is appropriate.

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GPU NUCLEAR

TMI
Emergency Procedure

Number: 1202-11

Title: **High Activity in Reactor Coolant**

Revision No.: 25

Applicability/Scope: TMI Division

Responsible Office: Plant Operations Director

Effective Date: 12/15/98

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	[illegible]	12-2-98
Procedure Owner	L.G. Noll for T.H. Goodbrake	12-10-98
PRG	[illegible]	12-10-98
Approver	L.G. Noll	12-10-98

1.0 SYMPTOMS

1. Alert Alarm on RC letdown radiation monitor RM-L1.

2. High Radiochemical analysis results (≥ 0.35 μCi/gm Dose Equivalent I-131 or > 100/E μCi/gm).

2.0 IMMEDIATE ACTION

A. Automatic Action

1. MU-V2A and MU-V2B close If a high alarm is received on RM-L1.

B. Manual Action

1. Announce on Page: "High Activity has been detected in letdown. All unnecessary personnel in the controlled area report to H.P. Checkpoint."

3.0 FOLLOWUP ACTION

Objective:

Determine activity in RCS, minimize radiation levels in the plant, minimize radioactive exposure to personnel, and to determine the cause for the high activity.

_____ 1. If RM-L1 is in high alarm, compare its reading to RM-L1 low. RM-L1 low should also have a high alarm.

If RM-L1 low is not tracking RM-L1.

Then have the Shift Supervisor evaluate restoring letdown per 1210.10.

_____ 2. If RM-L1 low is tracking RM-L1.

Then verify MU-V2A and MU-V2B are closed or close manually.

_____ 3. If letdown cannot be restored, then commence plant shutdown and cooldown to limit pressurizer level to ≤ 330 inches.

_____ 4. If pressurizer level reaches 380 inches while the reactor is critical, then trip the reactor.

_____ 5. Monitor RM-A5 channel for possible increase, if RM-A5 alert alarms and OTSG tube leak is suspected, refer to ATP 1210-5.

Form ES-401-6

Written Exam Question Worksheet

Q # 016

Examination Outline Cross-Reference RO SRO

KA # BW A02 AK2.2 Page 4.3-28 Tier # 1

RO/SRO Importance Rating 3.8 3.8 Group # 1

Measurement Knowledge of the interrelations between the Loss of NNI-X/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 ☑ SRO ☐ PRA Related **Correct Answer** B

Plant conditions:
- Reactor Power is 100%.
- The following ICS Stations are in HAND due to a BTU Limit modification in progress:
  FW-P-1A, FW-P-1B
  FW-V-16A, FW-V-16B
  FW-V-17A, FW-V-17B
  MAP alarm H-1-2, OTSG A BTU LIMIT, is actuated
  MAP alarm H-1-3, OTSG B BTU LIMIT, is actuated
- All other ICS stations are in AUTOMATIC

A Loss of ICS HAND Power occurs.

Identify the ONE (1) statement below that describes required operator action(s) for this situation.

A. Stabilize reactor power to match Main Feedwater flow.

B. Trip the Reactor, Turbine and both Main Feedwater Pumps.

C. Place both Feedwater Pumps and all Feedwater Valves into ICS Auto.

D. Transfer both Feedwater Pumps to the Motor Speed Changers in the Control Room.

Technical Reference EP 1202-41, Total or Partial Loss of ICS/NNI Hand Power, Rev. 31, Pages 3,4.

Open Exam Reference

Learning Objective V.D.21.03 State the Immediate Manual Actions required for a Total or Partial Loss of ICS/NNI Hand power per EP 1202-41.

Question Source ☐ New ☑ TMI Bank TMI Question #
☑ Modified TMI Bank Parent Question # 15

Question History ☐ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .7

Discriminant Validity Statements

A Plausible since this action would attempt to balance heat production with heat removal.

B Correct answer.

C Plausible since this would address the loss of ICS Hand power, but not the BTU limits.

D Plausible since this is an alternate method of Feedwater Pump control, but will not address the Feedwater valves.

Comments None.

CONTROLLED COPY

AmerGen	TMI - Unit 1 Emergency Procedure	Number 1202-41
Title **Total or Partial Loss of ICS/NNI Hand Power**		Revision No. 31
Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date 03/31/00

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	George J Dyler	3/29/200
Procedure Owner	[illegible]	3/29/2000
PRG	[illegible]	3/31/2000
Approver	[illegible]	3/30/00

3. If all ICS/NNI Stations are in "AUTO" no other actions occur.

4. The following ICS/NNI control stations will be affected only if they are in HAND, (Manual).

   a. FW-V-17A/B Fail to Mid Position.

   b. FW-V-16A/B Fail to Mid Position.

   c. FW-P-1A/B Fail to ~ 4100 rpm.

   d. MS-V-3A-F Fail to Mid Position.

   e. MU-V-17 Fails to Mid Position.

   f. MU-V-32 Fails to Mid Position.

   g. Pressurizer Heaters (Banks 1, 2 and 3) Fail to 50% Demand.

NOTE

The following ICS hardware is unaffected by a loss of 'Hand' power in any operating mode: "Unit Load Demand" control station, "Load Rate of Change" control station, "SG/Reactor" HAND/AUTO control station, "Main Turbine" HAND/AUTO control station, "SG A FW Demand" HAND/AUTO control station, "SG B FW Demand" HAND/AUTO control station, "SG A/B Load Ratio" HAND/AUTO control station, and "Reactor Demand" HAND/AUTO control station. The GE MARK-V main turbine generator control and the Diamond Control Rod station also are unaffected by only a loss of ICS/NNI "Hand" Power.

2.2 Manual Actions

CAUTION

Do not transfer ICS HAND/AUTO control stations to HAND. The associated demand will Fail to 50%.

CAUTION

Do not attempt to restore "HAND" power at this point. The demand associated with ICS final control elements that are under HAND control will go to an indeterminate value possibly resulting in undesirable main feedwater flow and/or OTSG pressure response.

1. Monitor feedwater flow and verify plant control is stable.

Form ES-401-6

Written Exam Question Worksheet

Q # 017

Examination Outline Cross-Reference RO SRO

KA # APE 003 AK2.05 Page 4.2-4 Tier # 1

RO/SRO Importance Rating 2.5 2.8 Group # 2

Measurement Knowledge of the interrelations between the Dropped Control Rod and the following: Control rod drive power supplies and logic circuits

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

The plant is at 100% power.

The following plant conditions exist:
- Reactor Protection System surveillance in progress on "C" RPS cabinet.
- "C" RPS cabinet's associated CRD breakers/electronic trips were tripped
- An AO has manually reclosed "C" RPS cabinets associated DC breakers
- The Fault Reset pushbutton has NOT been pushed on the CRD Diamond panel

If a loss of Vital Bus "D" were to occur at this time, which ONE of the following would be the immediate effect?

A. CRD Groups 1 and 3 would drop

B. CRD Groups 1, 2, 3 and 4 would drop

C. CRD Groups 5, 6 and 7 would drop

D. CRD Groups 1 through 7 would drop

Technical Reference OP 1105-9, Control Rod Drive System, Rev. 56 page 48
OPM Section F-02, Reactor Protection System, Rev. 9 page 116
OPM Section F-01, Control Rod Drive System, Rev. 7 page 58

Open Exam Reference

Learning Objective IV.E.13.07 Given a specific set of CRD breaker positions and electronic trip status, determine the positions of the control rods.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .7

Discriminant Validity Statements

A Plausible with a misunderstanding that the DC supply to only half of the safety rods would be lost.

B Plausible with a misunderstanding of which rods are powered from a DC supply and which are powered from an AC supply.

C Correct answer. All power would be lost to the regulating rods.

D Plausible with a misunderstanding that both the primary and secondary power supplies would be lost to all of the rods.

Comments None.

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GPU NUCLEAR

TMI
Operating Procedure

Number: 1105-9

Title: **Control Rod Drive System**

Revision No.: 56

Applicability/Scope: TMI Division

Responsible Office: Plant Operations Dir.

Effective Date: 11/25/98

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	Stuart Brantley	11-17-98
Procedure Owner	Stuart Brantley	11-17-98
PRG	[illegible]	11-19-98
Approver	L. G. Nora	11-18-98

5. CONTROL ON (Group 1 through 8 CRD Control Panel)

Indicates a particular group has been selected, either manually or due to the automatic sequencer, for control. Unless the CONTROL ON lamps at the PI Panel are also energized, the selected group (or rods within that group) will not respond to INSERT/WITHDRAW commands.

6. CONTROL ON lamps at the PI Panel

The CONTROL ON lamps on the PI Panel indicate that the rod has been transferred to the Aux Supply or that the associated group has been selected for control on its normal supply.

7. CLAMP/CLAMP REL

Used to energize and de-energize the clamping contactors which cross-connect the auxiliary supply with either the DC Hold or one of the 4 regularing supplies. Will not indicate clamp unless SY is indicated. White indicator lamps indicate the position of the alternate-action switch. Amber and green lamps indicate the clamping contactor as being energized and de-energized, respectively.

8. CRD TRAVEL IN/OUT

Indicates an insert or withdraw command has been directed to one of the safety or regulating groups. Will not indicate commands associated with Group 8.

9. Dilute Signal 1

Provided the safety groups are at OUT LIMIT, dilution control signal 1 is enabled if group 5 absolute position is less than 80 percent withdrawn. The 80 percent limit may be bypassed by the DILUTE SIGNAL keyswitch in System Logic Cabinet 2.

10. Dilute Signal 2

Dilute Signal 2 is enabled when Group 6 absolute position is above 95 percent and Group 6 relative is above 75 percent withdrawn. The No. 2 signal is disabled if either Group 6 absolute or relative position indications drop below 75 percent.

11. FAULT RESET

Resets a fault condition if fault has cleared. The faults may be MOTOR FAULT, PROGRAMMER LAMP FAULT A and B, MOTOR POWER SUPPLIES, and SYSTEM POWER SUPPLIES A and B. Also resets the electronic trips when groups 1-7 are not at their IN Limit (This is necessary to reset these trips after RPS testing). Lights only when depressed.

12. FAULT lamp on the PI Panel.

The FAULT lamp indicates a 7 inch misalignment of that rod from the group average position.

FOR INFORMATION ONLY

SECTION F-02
REVISION 9

TMI OPERATIONS PLANT MANUAL
REACTOR PROTECTION SYSTEM

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Concur

Owner — Stuart Brantley — Date 11-19-99

Operations Training Manager, TMI — R L Campbell — Date 12/1/99

Training Coordinator, TMI — Linda L. Bitter — Date 12-2-99

Approved

Plant Operations Director TMI — D W Cook — Date 12/3/99

7.4 RPS Test Circuits and Trip Relay Operation (refer to Figures 61, 62)

Each RPS main trip relay controls a single contact in the supply and return power lines for the Control Rod Drive (CRD) power supply breakers undervoltage coils (Refer to CRD section of training manual). The trip scheme for the CRD undervoltage coils is set up in such a way (refer to Figure 62) that when 2 out of 4 RPS master trip relays are de-energized the contacts associated with those RPS cabinets open in each CRD undervoltage coil supply and return power lines. The CRD undervoltage coils then deenergize tripping all CRD power supply breakers, the segment arms in the Control Rod Drives release, the roller nuts disengage from the lead screw and control rods fall into the core.

Test circuitry is set up in the RPS cabinets so a single CRD breaker or breaker set (in the case of the CRD DC hold breakers) can be test tripped from its associated RPS cabinet through the use of simulate trip toggle switches on the Rx trip module in each RPS cabinet (see Figure 62). The simulate trip toggles switches will only open contacts for the CRD breaker or breaker sets UV coil associated with that RPS cabinet and will not cause a reactor trip. The manual trip scheme (use of 2 or more out of 4 simulate trip toggle switches) is set up so the CRD breakers can be tested via following scheme.

RPS CABINET	BKR(s) Tested from that Cabinet
A	CB 10
B	CB 11
C	CB1, CB2
D	CB3, CB4

When any of the test signals are applied, (i.e., a module taken to the "test" position) the master trip relay is de-energized and that cabinet's contact opens in the power lines of the CRD undervoltage coils. The test lamp on the test module in the RPS cabinet becomes bright and the protective subsystem lights for that cabinet will also become bright.

This will also occur if any modules shown in Figure 62 are withdrawn from the cabinet. This will place the RPS in a 1 out of 3 trip scheme. That is to say if one more of the remaining three untripped channels were to trip, the contact scheme for de-energizing the CRD UV coils would be satisfied and a reactor trip would be initiated.

FOR INFORMATION ONLY

SECTION F-01
REVISION 7

TMI OPERATIONS PLANT MANUAL

CONTROL ROD DRIVE SYSTEM (ELECTRICAL AND MECHANICAL)

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Concur

Owner — Date 12-4-98

Operations Training Manager, TMI — Date 12/9/98

Training Coordinator, TMI — Date 12-11-98

Approved

Plant Operations Director TMI — Date 12-15-98

11.0 REACTOR TRIP

11.1 General Description

Each power supply has a separate, redundant, identical input. Each of the programmers is also designed to be redundant. The four DC breakers are also wired in redundant sets of two each. This redundancy satisfies the safety trip requirements as provided from the Reactor Protection System (RPS) trip logic. Each of four RPS trip channels feeds four Reactor Trip (RT) modules. One main AC breaker is associated with RT module A and the other with RT module B. One set of DC breakers and one half of the redundant programmers are associated with RT module C; the other set of DC breakers and the other half of the redundant programmers are controlled by RT module D. Any combination of 2-out-of-4 RPS trip channels will cause each RT module to activate and its associated breaker/programmer to open. By combining redundant devices to trip the drives, protection against a single failure is realized.

a. Main AC Breakers - A and B:

❶ The breaker is tripped either by a manual trip or by the RPS trip logic.

❷ Each breaker has source interruption devices:

a) Overvoltage - Greater than 140V AC sensed at the transformer output will trip that side breaker.

b) Undervoltage - Less than 75V sensed at the output of DC hold rectifying banks. Loss of both A and CC phases will trip both the A and B breakers. This feature provides ratchet trip protection.

b. The DC hold bus breakers (breakers C and D) are tripped by either a manual trip or the RPS trip logic.

c. Associated with each programmer are electronic trips. These trips are actuated by either a manual trip or the RPS. There are two contacts per programmed power supply. Opening a contact interrupts 24V DC power to one light source in a power supply. If both contacts open, both light sources turn off; thus, no gating signal is sent to the SCRs, and all rods powered by that supply will trip. The trip contacts located in the main lamp power supply for all programmers are designated as the "E" electronic trips. The trip contacts located in the secondary lamp power supply for all programmers are designated the "F" electronic trips.

d. There is also a Diverse Scram System which will trip breakers 1G-2A and 1L-2A. This trip can be manually actuated from the Control Room or automatically actuated if RC-PT-963 and RC-PT-949 detect greater than 2500 psig RCS pressure. See OPM Section F-2 (Reactor Protection System) for a detailed DSS description.

e. To trip all rods, ONE of the following breaker/electronic trip combinations must exist:

❶ A and B breakers open.
❷ A and D breakers open plus "F" electronic trips.
❸ B and C breakers open plus "E" electronic trips.
❹ C and D breakers open plus "E" and "F" electronic trips.
❺ 1G-2A and B breakers open.
❻ 1G-2A and D breakers open plus "F" electronic trips.
❼ 1L-2A and A breakers open.

Form ES-401-6

Written Exam Question Worksheet

Q # 018

Examination Outline Cross-Reference RO SRO

KA # BW E02 EA2.2 Page 4.3-3 Tier # 1

RO/SRO Importance Rating 3.2 3.8 Group # 2

Measurement Ability to determine and interpret the following as they apply to the VSSV/Post-Trip Stabilization: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

A reactor trip occurred during performance of maintenance on the RPS cabinets. Performance of ATP 1210-1 is in progress and the following plant conditions exist:

- RCS pressure is 1800 psig and decreasing at 50 psig every 2 minutes
- OTSG A SU level is 40 inches and decreasing at 2 inches per minute
- OTSG B OP level is 95% and rising at 5% per minute
- Makeup flow is 350 gpm
- Pressurizer level is 80 inches and dropping 3 inches per minute
- SCM is 32 degrees F and dropping 1 degree per minute
- MUT level is 78 inches and dropping 10 inches per minute

Which ONE of the following identifies the most immediate operator action required for these conditions?

A. Trip all RCPs

B. Trip both MFW pumps

C. Initiate both trains of HPI

D. Open MU-V-14A or MU-V-14B, BWST suctions to the MUPs

Technical Reference ATOG 1210-1, Rev. 40 page 3.

Open Exam Reference

Learning Objective V.E.01.03 Given the inability to accomplish any action/verification, list the action(s) which must be taken. (IF/THEN Statements)

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge

☑ Comprehension or Analysis

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5 ☑ 55.45 .13

Discriminant Validity Statements

A Plausible is SCM continues to decrease below 25 degrees.

B Correct answer.

C Plausible if Pressurizer level drops below 20 inches or SCM is lost.

D Plausible if MUT level continues to decrease below 55 inches.

Comments None.

CONTROLLED COPY

GPU NUCLEAR	TMI Abnormal Transient Procedure	Number 1210-1
Title **Reactor Trip**		Revision No. 40
Applicability/Scope TMI Division	Responsible Office Plant Operations Dir.	Effective Date 10/19/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator		10/15/99
Procedure Owner	James N Faulls	10/15/99
PRG	John S. DeBorb	10/18/99
Approver	James N Faulls	10/15/99

2.3 Control Main Feedwater.

Verify FW-V-5A/B stroking closed or are closed; AND ICS controlling MFW flow.

IF ICS is not controlling MFW flow;	THEN take Hand control of the FW regulating valves and/or MFW pumps AND control proper OTSG level and valve DP.
IF MFW flow is still excessive;	THEN trip both MFW pumps, **PRIOR TO** exceeding 97.5% OTSG level.

2.4 Verify ICS/NNI Power On (PCL).

IF both AUTO AND HAND power lights are off;	THEN trip MFW pumps, AND verify EFW, AND maintain primary to secondary heat transfer by using the ADV B/U loaders to control OTSG pressure. AND Refer to EP 1202-40.
IF any other lights or combination of lights are off;	THEN refer to EP 1202-40/41/42.

2.5 Verify 4160 volt buses 1A, 1B, 1C, 1D and 1E energized (CR and PR).

IF loss of station power has occurred;	THEN verify starting and loading of at least one D/G. AND Restore Make-up, seal injection and EFW. AND Refer to EP 1202-2.
IF both ES buses are deenergized;	THEN isolate letdown AND seal return, AND verify steam driven EFW pump operating AND Refer to EP 1202-2.

2.6 Start 2nd Make-up Pump and open MU-V-217.

IF MU Tank is less than 55";	THEN open MU-V-14A or B as necessary to maintain MU Tank level greater than 55".
IF unable to maintain pressurizer level greater than 20 inches;	THEN initiate HPI.

Form ES-401-6

Written Exam Question Worksheet

Q # 019

Examination Outline Cross-Reference RO SRO

KA # BW A01 AA1.3 Page 4.3-27 Tier # 1

RO/SRO Importance Rating 3.7 3.7 Group # 2

Measurement Ability to operate/monitor the following as they apply to the Runback: Desired operating results during abnormal/emergency situations

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

Initial conditions:
- Plant at 80% power during a power escalation.
- Rod index is 275
- FW-P-1A and FW-P-1B in operation

The following sequence of events occurred:
- Loss of FW-P-1A
- Runback initiated in response to the loss of FW-P-1A
- 3 seconds after the runback initiated, a rod in Group 5 drops to the bottom of the core.

Which ONE (1) of the following describes the plant response to these conditions?

A. Runback continues at 30% per minute to 482 MWd

B. Runback continues at 30% per minute to 585 MWd

C. Runback continues at 50% per minute to 482 MWd

D. Runback continues at 50% per minute to 585 MWd

Technical Reference ARP MAP H-1-1, page 1 Rev. 20.

Open Exam Reference

Learning Objective IV.E.27.60 Given a set of plant conditions, determine the correct ICS response and any actions that must be taken if the ICS response is incorrect.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .5, .6

Discriminant Validity Statements

A Correct answer.

B Plausible if runback rate determination and setpoint hierarchy are misinterpreted.

C Plausible if runback rate determination and setpoint hierarchy are misinterpreted.

D Plausible if runback rate determination and setpoint hierarchy are misinterpreted.

Comments None.

AmerGen	TMI - Unit 1 Alarm Response Procedure	Number **MAP H**
Title **Main Annunciator Panel H**		Revision No. **66**
Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date **03/23/00**

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

H-1-1 **Rev. 20** **03/23/00**	**H-1-2** **Rev. 20** **03/23/00**	**H-1-3** **Rev. 9** **03/23/00**	**H-1-4** **Rev. 13** **03/23/00**	H-1-5 Rev. 12 03/23/00	**H-1-6** **Rev. 18** **03/23/00**	**H-1-7** **Rev. 18** **03/23/00**	**H-1-8** **Rev. 13** **03/23/00**
H-2-1 **Rev. 14** **03/23/00**	**H-2-2** **Rev. 14** **03/23/00**	**H-2-3** **Rev. 12** **03/23/00**	**H-2-4** **Rev. 10** **03/23/00**	H-2-5 Rev. 11 03/23/00	H-2-6 Cancelled 10/16/91	H-2-7 Cancelled 10/16/91	**H-2-8** **Rev. 9** **03/23/00**
H-3-1 Cancelled 10/16/91	**H-3-2** **Rev. 16** **03/23/00**	**H-3-3** **Rev. 13** **03/23/00**	**H-3-4** **Rev. 10** **03/23/00**	H-3-5 Rev. 8 03/23/00	H-3-6 Cancelled 10/16/91	**H-3-7** **Rev. 9** **03/23/00**	**H-3-8** **Rev. 10** **03/23/00**

	Signature	Date
Originator	George J Butler	03/03/2000
Originator	[illegible]	3/14/2000
Procedure Owner	[illegible]	3/15/2000
G	[illegible]	3/16/2000
Approver	[illegible]	3-15-00

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TRAINING DEPT

ALARM:

ICS RUNBACK

SET POINTS:

Loss of one RCP and > 75 percent power
Loss of two RCP and > 55 percent power
Loss of one main feed pump and > 60 percent power
RC flow limit percent core power $\geq \frac{105(\text{RC flow})}{131.32 \times 10^{6} \text{ lbs/hr}}$
Asymmetric rod 9" out of alignment with its group average rod position and > 60 percent power.

CAUSES:

RCP Trip
MFP Trip
Dropped Rod

AUTOMATIC ACTION:

Unit runs back to:
75 percent power on a loss of (1) RCP at a rate of 50 percent/min. ≈ 666 MWe
50 percent power on a loss of (2) RCP at a rate of 50 percent/min. ≈ 435 MWe
68 percent power on a loss of (1) MFP at a rate of 50 percent/min. ≈ 585 MWe
The RC flow limit at a rate of 20 percent min.
55 percent power on a Assy. rod fault at a rate of 30 percent/min. ≈ 482 MWe

NOTE

Runbacks are based on a MWe. Input power may vary due to plant efficiencies.

Form ES-401-6

Written Exam Question Worksheet

Q # 020

Examination Outline Cross-Reference RO SRO

KA # BW A04 AK1.1 Page 4.3-32 Tier # 1

RO/SRO Importance Rating 3.0 3.3 Group # 2

Measurement Knowledge of the operational implications of the following concepts as they apply to the Turbine Trip: Components, capacity, and functions of emergency systems.

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

Which ONE (1) of the following is the reason for the Anticipatory Reactor Trip on a Turbine Trip above 45% power?

A. Prevent challenges to the PORV due to RCS pressure increase.

B. Prevent excessive core outlet temperatures due to the RCS heatup.

C. This is the maximum capacity of the Main Steam Safety Valves.

D. This is the maximum capacity of the Turbine Bypass and Atmospheric Dump Valves.

Technical Reference OPM Section F-02, Reactor Protection System, Rev. 9 page 19

Open Exam Reference

Learning Objective IV.E.14.06 List the automatic reactor trips including setpoints and state the bases for each as found in Tech Specs.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .8, .10 ☐ 55.43 ☑ 55.45 .3

Discriminant Validity Statements

A Correct answer:

B Plausible due to resultant RCS temperature increase during transient.

C Plausible in the examinee does not the the MSSV relief capacity.

D Plausible in the examinee does not the the TBV and ADV relief capacity.

Comments None.

FOR INFORMATION ONLY

TMI OPERATIONS PLANT MANUAL

REACTOR PROTECTION SYSTEM

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Concur

Owner Stuart Brontley Date 11-19-99

Operations Training Manager, TMI R L Lamb Date 12/1/99

Training Coordinator, TMI Linda L. Butler Date 12-2-99

Approved

Plant Operations Director TMI D W [illegible] Date 12/3/99

The feedwater pumps tripped bistable shall be set to bypass at reactor power levels below 7% and to remove the bypass at reactor power levels above 7% or as documented in plant operating procedures.

The main turbine tripped bistable shall be set to bypass at reactor power levels below 45% and to remove the bypass at reactor power levels about 45% or as documented in plant operating procedures.

The pressure switches on the feedwater pumps shall be set to open at 75 psig on decreasing pressure and to reset (close) on increasing pressure at 95 psig.

The pressure switches on the main turbine hydraulic console shall be set to open on decreasing pressure at 800 psig and to reset (close) at 1000 psig.

Operation

During startup and shutdown of the turbine and feedwater pumps plant operating procedures must be followed to preclude inadvertent reactor trips caused by the pressure switches. Setpoints and limitations are listed in plant setpoint procedure.

Startup - During startup conditions the turbine and feedwater pumps oil hydraulic system must be on and at pressure prior to reaching reactor power level.

Normal operation - The turbine and feedwater oil hydraulic system must maintain pressure to prevent reactor trips.

Shutdown - During shutdown conditions the turbine and feedwater pumps oil hydraulic system pressures must be maintained until the reactor power is below the bypass setpoints.

Casualty Events

A turbine trip or feedwater pump trip or loss of pressure to either oil hydraulic system is deemed a casualty event for which this anticipatory trip is provided to prevent excessive high reactor coolant pressure transients, thereby preventing challenges to the PORV and code safeties.

Maintenance

Test switches are provided on the contact buffer modules that are located in the RPS and are used for testing the function of the contact buffer. Testing of the bistables can be accomplished using the power range test circuit. Testing of the RPS modules should coincide with scheduled maintenance of the RPS channel.

Redundant channels allow testing and maintenance during any plant operating mode.

4.15 Tech Spec Requirement/Surveillance Testing

The Technical Specifications require that a minimum number of channels be available during operation. Tech Spec 3.5.1, Table 3.5-1 refers to channel operability and degree of redundancy required. Refer to Tech Spec 4.1.1 for testing requirements to determine channel operability and calibration accuracy. Table 2 is a listing of the Technical Specification Surveillance Procedures which are used to satisfy these requirements.

Form ES-401-6

Written Exam Question Worksheet

Q # 021

Examination Outline Cross-Reference RO SRO

KA # EPE 009 EA1.14 Page 4.1-5 Tier # 1

RO/SRO Importance Rating 3.4 3.4 Group # 2

Measurement Ability to operate and/or monitor the following as they apply to a small break LOCA: Secondary pressure control

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** A

The following plant conditions exist:
- Small Break LOCA cooldown in progress
- cooldown rate is 30°F per hour
- incore thermocouple temperature is 456°F

ATP 1210-6, Small Break LOCA Cooldown, requires you to "Maintain OTSG pressure so that secondary T-sat is 40 to 60°F lower than incore thermocouple temperature."

Which ONE (1) of the following would be the correct OTSG pressure to accomplish this step?

A. 255 psig

B. 295 psig

C. 435 psig

D. 685 psig

Technical Reference ATOG 1210-6, Small Break LOCA Cooldown, Rev.28 page 3

Open Exam Reference

Learning Objective V.E.06.10 Given plant conditions determine if entry into ATP-1210-6, Small Break LOCA Cooldown, is required and what actions are to be taken.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .5, .6

Discriminant Validity Statements

A Correct answer (48°F lower).

B Plausible, but neglects conversion between psia and psig

C Plausible, but is P-sat for the current incore temperature

D Plausible, but 47°F higher

Comments None.

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GPU NUCLEAR	TMI Abnormal Transient Procedure	Number 1210-6
Title: **Small Break LOCA Cooldown**		Revision No. 28
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: 10/15/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	[illegible]	10-14-99
Procedure Owner	James R Powles	10/15/99
PRG	[illegible]	10/15/99
Approver	James R Powles	10/15/99

[ ] 2.8 IF AT ANY TIME CORE COOLDOWN RATE < 40°F/hr; THEN perform the following actions:

> **CAUTION**
>
> If the HSPS Low Pressure Isolation is NOT bypassed, lowering OTSG pressure below 600 psig could cause a loss of heat sink when MFW is isolated.

____ 2.8.1 Maintain OTSG pressure so that secondary T_{sat} is 40 to 60°F lower than incore thermocouple temperature. Maintain OTSG level.

> **NOTE**
>
> RCITS hot leg level instrumentation is NOT functional with hot leg vents open.

____ 2.8.2 Open hot leg High Point Vents in loop(s) with no operating RC Pumps.

____ RC-V-40A
____ RC-V-41A

____ RC-V-40B
____ RC-V-41B

> **CAUTION**
>
> Do not go below steam pressure required for EF-P-1 (150 psig) unless EF-P-2A OR EF-P-2B OR auxiliary steam (for EF-P-1) is available.

2.8.3 IF core cooldown rate is still < 40°F/hr; THEN
1) Maintain OTSG pressure so that secondary T_{sat} is 90 to 100°F lower than incore thermocouple temperature.
2) Raise and maintain OTSG level at 75-85%.

____ 2.8.4 IF primary to secondary heat transfer cannot be established AND RCS cooldown rate is < 40°F/hr AND at least one HPI pump injecting water into the RCS; THEN open the PORV and PORV Block, UNTIL primary to secondary heat transfer has been established OR LPI > 1000 gpm per leg.

2.9 Verify containment cooling as follows:

____ 2.9.1 Run all available AH-E-1s (using SLOW speed if RB 4 psig ESAS actuation has occurred).

____ 2.9.2 Verify RR system cooler outlet pressure is 55-60 psig as read on PI-224/225/226.

Form ES-401-6

Written Exam Question Worksheet

Q # 022

Examination Outline Cross-Reference RO SRO

KA # BW E08 2.4.4 Page 2-11 Tier # 1

RO/SRO Importance Rating 4.0 4.3 Group # 2

Measurement LOCA Cooldown: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency/abnormal procedures

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** C

A Small Break LOCA Cooldown is in progress in accordance with ATP 1210-6.

The following plant conditions exist:
- all four (4) Reactor Coolant Pumps are secured
- RCS pressure is 385 psig and stable
- both OTSG levels are 63% and being raised to 75 - 85%
- both OTSG pressures are approximately 145 psig and stable
- both T-hots are approximately 440°F
- both T-colds are approximately 400°F
- the five highest incore thermocouples average 490°F

Which ONE (1) of the following procedure transitions should take place?

A. Go To ATP 1210-4, Lack of Primary to Secondary Heat Transfer.

B. Go To ATP 1210-7, Large Break LOCA Cooldown.

C. Go To ATP 1210-8, RCS Superheated.

D. Go To ATP 1210-9, HPI Cooling - Recovery from Solid Operations.

Technical Reference ATOG 1210-6, Small Break LOCA Cooldown, Rev. 28 page 2

Open Exam Reference Steam Tables

Learning Objective V.E.08.09 Given plant conditions determine if the RCS is superheated and what actions are required.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .10 ☐ 55.43 ☑ 55.45 .2, .6

Discriminant Validity Statements

A Plausible because a lack of primary to secondary heat transfer is occurring.

B Plausible because core flood tanks should be dumping.

C Correct answer.

D Plausible because HPI cooling may be in progress, but 1210-8 would be a higher priority.

Comments None.

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GPU NUCLEAR	TMI Abnormal Transient Procedure	Number 1210-6
Title: **Small Break LOCA Cooldown**		Revision No. 28
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: 10/15/99

This document is within QA plan scope: [X] Yes [ ] No

Safety Reviews Required: [X] Yes [ ] No

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	Signature	Date
Originator	[illegible]	10-14-99
Procedure Owner	James R Paules	10/15/99
PRG	[illegible]	10/15/99
Approver	James R Paules	10/15/99

1.0 **IMMEDIATE ACTIONS**

None

2.0 **FOLLOW-UP ACTIONS**

CAUTION

1. Do not start any 4KV or 6.9KV motors during block loading.
2. Do not attempt to operate ES valves to the non-ES position until the ES signal is bypassed.
3. Do not trip 480V or 4160V ES breakers until the ES signal is bypassed. The breaker anti-pump feature prevents reclosure of tripped breakers until the start signal is cleared.

_____ 2.1 Verify that loss of 25°F subcooled margin is being treated.

_____ 2.2 Verify that all Reactor Trip Isolation valves have closed. .

2.3 Verify Safety System Status

_____ IF AT ANY TIME an ES actuation setpoint has been reached; THEN verify all components have been actuated for the respective ES signal(s).

_____ 2.4 Refer to Attachment 1 and implement EPIP-TMI-.01, EMERGENCY CLASSIFICATION, and additional implementing procedures as required.

_____ 2.5 Maintain OTSG pressure so that secondary T_{sat} is lower than incore thermocouple temperature. Maintain OTSG level.

_____ 2.6 IF AT ANY TIME **SUPERHEAT** exists; THEN go to ATP 1210-8.

_____ 2.7 IF AT ANY TIME a **OTSG TUBE LEAK/ RUPTURE** has occurred; THEN refer to ATP 1210-5.

Form ES-401-6

Written Exam Question Worksheet

Q # 023

Examination Outline Cross-Reference RO SRO

KA # APE 025 AA2.01 Page Tier # 1

RO/SRO Importance Rating 2.7 2.9 Group # 2

Measurement Ability to determine and interpret the following as they apply to the Loss of RHR System: Proper amperage of running LPI/DHR pumps

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** C

Current plant conditions:
- The plant is in Refueling Shutdown
- "A" train Decay Heat Removal is in service
- The Reactor Coolant System is being drained down for RV Head removal
- The "B" train Decay Heat Removal is being lined up for operation

Which ONE of the following would be a reason for DH-P-1B developing fluctuating motor amps when it is started?

A. DH-V-4B closed

B. DH-V-5B closed

C. DH-V-12B closed

D. DH-V-38B closed

Technical Reference EP 1202-35, Loss of Decay Heat Removal System, Rev. 33 page 4

Open Exam Reference

Learning Objective 000025050 1 During Decay Heat Removal operations, respond to a loss of decay heat removal IAW EP 1202-35, Loss Of DHR System.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5 ☑ 55.45 .13

Discriminant Validity Statements

A Plausible because this would isolate the discharge path, but would not cause pump heating leading to cavitation due to the recirculation path through the Decay Heat Removal Cooler.

B Plausible because this would isolate the suction flowpath during the Low Pressure Injection Mode, but is not in the flowpath for Decay Heat Removal

C Correct answer. This would isolate the suction path.

D Plausible because this would isolate part of the discharge path, but would not cause pump heating leading to cavitation due to the recirculation path through the Decay Heat Removal Cooler.

Comments None.

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GPU NUCLEAR

	TMI Emergency Procedure	Number **1202-35**
Title **Loss of Decay Heat Removal System**		Revision No. **33**
Applicability/Scope TMI Division	Responsible Office Plant Operations Dir.	Effective Date 02/26/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

SIMULATOR ROOM FILE COPY

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	Signature	Date
Originator	Stuart Brantley	Feb 18, 1999
Procedure Owner	Stuart Brantley	Feb 18, 1999
PRG	[illegible]	2-19-99
pprover	[illegible]	2-18-99

1.0 **SYMPTOMS**

1. Increasing Reactor Coolant Temperature (indicates greater than 140°F as read on DH-6-TI-1/2 Console Center and Incore thermocouple temperatures increasing.
2. D.H. Pump Flow Lo (Annunciator C-1-7, and low flow indicated on DH-F1-802A or DH-F1-803A)
3. D.H. Out D.H. Removal Cooler A/B Temp. Hi (Computer points at 200°F A0933/A0935)
4. RCS Draindown Lvl Hi/Lo, (C-1-5), and Computer Points A0952/A0953
5. 4 KV E.S. Motor Trip (Annunciator B-1-2, on overcurrent and undervoltage, and Computer Points L2165/L2167, L2909/L2910) (DH-P-1A(B) Motor Status Computer Points L2829/L2831)
6. D.H. Removal System Valve Closed Suction or Partially Closed as indicated by console indication.
7. Increasing RCS pressure
8. Unstable discharge pressure for DH-P-1A(B)
9. Unstable motor amps for DH-P-1A(B)

.0 **IMMEDIATE ACTION**

A. Automatic Action

1. None

B. Manual Action

1. If DH-P-1A(B) has inadequate suction head, trip DH-P-1A(B) and do not restart until appropriate follow-up actions have been completed.
2. Verify DH-V-1, DH-V-2, DH-V-3, and DH-V-4A(B) are open.
3. Verify DC-P-1A(B) and DR-P-1A(B) are running in the loop that is (was) in service.
4. Verify RC System water level and stop all operations that would decrease RCS inventory.

Form ES-401-6

Written Exam Question Worksheet

Q # 024

Examination Outline Cross-Reference RO SRO

KA # EPE 029 EA1.08 Page 4.1-9 Tier # 1

RO/SRO Importance Rating 4.5 4.5 Group # 2

Measurement Ability to operate and/or monitor the following as they apply to a ATWS: Reactor trip switch pushbutton

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

Plant conditions:
- ATWS in progress
- The primary CRO depresses the Manual Reactor Trip pushbutton.

Which ONE (1) of the following sets of conditions would positively ensure that the operator's action was successful AND that ALL CRDMs had power removed?

A. RPS Breaker Trip Lights on ALL 4 RPS Cabinets are bright.

B. RPS Protective Subsystem Lights on ALL 4 RPS Cabinets are bright.

C. All IN-LIMIT lamps on the Diamond Control Panel for Groups 1-7 drop are lit.

D. Alarms G-1-3 (CRD DC Trip Brkr Open) AND G-1-4 (CRD AC Trip Brkr Open) actuate.

Technical Reference OPM Section F-02, Reactor Protection System, Rev. 8, Page 117.

Open Exam Reference

Learning Objective IV.E.14.10 Given the RPS cabinet status and/or CRD Breaker status, determine if a reactor trip has occurred and/or which control rod groups have tripped.

Question Source ☐ New ☑ TMI Bank TMI Question #
☑ Modified TMI Bank Parent Question # 17

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .5, .6

Discriminant Validity Statements

A Correct answer.

B Plausible misconception that the RPS Protective Trip lights indicate the associated breaker has opened. These lights energize when the RPS Subsystem Reactor Trip module actuates - rather than actual operation (opening) of CRD breakers.

C Plausible misconception that all associated rods are required to be below the in-limit in order for the lamps to be energized.

D Plausible misconception that this alarm combination confirms that all breakers associated with each alarm have opened. Alarm actuation logic does not require ALL breakers to be open.

Comments None.

FOR INFORMATION ONLY

SECTION F-02
REVISION 9

TMI OPERATIONS PLANT MANUAL
REACTOR PROTECTION SYSTEM

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Concur

Owner — Stuart Bromley — Date 11-19-99

Operations Training Manager, TMI — R L Campbell — Date 12/1/99

Training Coordinator, TMI — Linda L. Butler — Date 12-2-99

Approved

Plant Operations Director TMI — D W Rich — Date 12/3/99

Each RPS cabinet has indicating lights on the outside top to tell the operator that status of each cabinet (see Figure 61). The turbine trip and FW trip bypass lights are bright when the bypass bistables are energized (trip function bypassed). Fan failure lights will be bright when there is a loss of power to the cabinet fans or low flow as sensed by a ΔP switch. The Protective Subsystem lights come on bright when a channel trips due to a setpoint being exceeded, a test trip signal is inserted or a module withdrawn or taken to test, or when a simulate test toggle switch in that RPS cabinet is taken to trip. The Protective Subsystems lights are set up such that when a channel trips light #1 signifies A channel tripped, light #2 signifies B channel tripped, light #3 signifies channel C and light #4 signifies channel D. When a test signal is inserted the respective light for that cabinet will become bright in the numbering sequence described above on both outside top of cabinet and inside on the test module. When a channel is tripped or in test the respective Protective Subsystem light for that channel will be bright on all RPS cabinets (i.e., if "A" channel trips then light #1 on all RPS cabinets becomes bright). However, when the simulate trip toggle switches are used during CRD breaker testing the only lights that become bright are the lights above the toggle switch and the lights on the outside top of that RPS cabinet. The Manual Bypass light becomes bright when the channel is placed in manual or shutdown bypass and when the channel is placed in manual bypass a light on the test module (Figure 61) also becomes bright. When the shutdown bypass keyswitch is actuated not only does the outside top light become bright but a module in the right side of the RPS cabinet will have bright lights indicating the relay has picked up, the operators overhead alarm has actuated and the light above the cabinet has become bright. The breaker trip light will become bright when the CRD breaker associated with that cabinet is tripped.

When an RPS channel trips due to exceeding a setpoint or putting a module in test the channel is reset by first clearing the trip condition (or taking the test module selector switch to "operate"), then resetting the initiating module by using a reset spring return toggle switch on the relay status module shown in Figure 61. When the master trip relay is de-energized due to exceeding a limiting safety system setpoint the subsystem trip light on the relay module becomes bright.

The transfer circuitry for the non-nuclear instrument inputs to the ICS has been removed from the "A" RPS cabinet. Transfer of neutron power, RCS flow, and RCS pressure from "A" to "B" RPS cabinet input is accomplished by the SASS pushbuttons on console center. SASS operation is explained in section F-3 of the Operation Plant Manual.

Form ES-401-6

Written Exam Question Worksheet

Q # 025

Examination Outline Cross-Reference RO SRO

KA # APE 032 AK3.01 Page 4.2-24 Tier # 1

RO/SRO Importance Rating 3.2 3.6 Group # 2

Measurement Knowledge of the reasons for the following responses as they apply to the Loss of SR NI: Startup termination on source-range loss

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

Plant startup is in progress. Source Range channels indicate 100 cps. Group 5 rods are being withdrawn from 75% to 100% when Source Range Channel NI-11 fails low.

Based on this condition, the startup:

A. Should be terminated AND verify the core is 1% deltaK/K shutdown.

B. Should be terminated AND Group 5 rods fully inserted.

C. May continue provided NI-12 is operational.

D. May continue provided NI-3 is operational.

Technical Reference T.S 3.5.1 pages 3-27 and 3-29, Amendment 189.

Open Exam Reference

Learning Objective IV.E.11.15 State or identify the Nuclear Instrumentation channel operability requirements for a plant startup and power operation as delineated in Tech Specs section 3.5.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5, .10 ☐ 55.43 ☑ 55.45 .6, .13

Discriminant Validity Statements

A Plausible if it is determined to terminate the startup, 1% deltak/k shutdown would be a required condition.

B Plausible since this would have all controlling groups inserted back to the original position prior to initiating startup.

C Correct answer.

D Plausible since this instrument would be required for continuing the startup.

Comments None.

4-25

3.5 INSTRUMENTATION SYSTEMS

3.5.1 OPERATIONAL SAFETY INSTRUMENTATION

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5-1, Column "A" and "B" are met, except as provided in Table 3.5-1, Column "C". Specification 3.0.1 applies.

3.5.1.2 The key operated channel bypass switch associated with each reactor protection channel may be used to lock the reactor trip module in the untripped state as indicated by a light. Only one channel shall be locked in this untripped state at any one time. Unit operation at rated power shall be permitted to continue with Table 3.5-1, Column "A". Only one channel bypass key shall be kept in the control room.

3.5.1.3 In the event the number of protection channels operable falls below the limit given under Table 3.5-1, Column "A", operation shall be limited as specified in Column "C". Specification 3.0.1 applies.

3.5.1.4 The key operated shutdown bypass switch associated with each reactor protection channel shall not be used during reactor power operation (except for required maintenance or testing).

3.5.1.5 During START-UP when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade.

3.5.1.6 During START-UP, HOT STANDBY or POWER OPERATION, in the event that a control rod drive trip breaker is inoperable, within one hour place the breaker in trip. Specification 3.0.1 applies.

3.5.1.7 During START-UP, HOT STANDBY or POWER OPERATION, in the event that one of the control rod drive trip breaker diverse trip features (shunt trip or undervoltage trip attachment) is inoperable:

a. Restore to OPERABLE status within 48 hours or

b. Within one additional hour place the breaker in trip.

Specification 3.0.1 applies.

TABLE 3.5-1

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot be Met
A. Reactor Protection System			
1. Manual pushbutton	1	0	(a)
2. Power range instrument channel	2	1	(a)
3. Intermediate range instrument channels	1	0	(a) (b)
4. Source range instrument channels	1	0	(a) (c)
5. Reactor coolant temperature instrument channels	2	1	(a)
6. (Deleted)	--	--	- - - -
7. Flux/imbalance/flow	2	1	(a)
8. Reactor coolant pressure			
a. High reactor coolant pressure instrument channels	2	1	(a)
b. Low reactor coolant pressure instrument channels	2	1	(a)

Form ES-401-6

Written Exam Question Worksheet

Q # 026

Examination Outline Cross-Reference RO SRO

KA # APE 033 AK1.01 Page 4.2-26 Tier # 1

RO/SRO Importance Rating 2.7 3.0 Group # 2

Measurement Knowledge of the operational implications of the following concepts as they apply to Loss of IR NI: Effects of voltage changes on performance

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

The compensating voltage power supply for NI-3 fails to zero voltage output. In comparison with NI-4, which is operating properly, NI-3 will read

A. significantly higher during startup, and have negligible error at 100% power.

B. significantly lower during startup, and have negligible error at 100% power.

C. significantly higher from startup through 100% power.

D. significantly lower from startup through 100% power.

Technical Reference OPM Section F-4, Nuclear Instrumentation System, Rev. 8 Pages 17-20 & 22

Open Exam Reference

Learning Objective IV.E.11.09 Describe or identify the affects of improper discriminator or compensating voltage settings on the source and intermediate range Nuclear Instrumentation channels.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .8, .10 ☐ 55.43 ☑ 55.45 .3

Discriminant Validity Statements

A Correct answer

B Plausible with the misunderstanding of the effects of under-compersation.

C Plausible with the misunderstanding of neutron flux overwhelming gamma flux and gamma linearity with power at higher power levels.

D Plausible with the misunderstanding of both B and C above.

Comments None.

FOR INFORMATION ONLY

TMI OPERATIONS PLANT MANUAL
NUCLEAR INSTRUMENTATION SYSTEM

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Concur

Owner _____ Date 8-9-96

Operations Training Manager, TMI _____ Date 8-16-96

Training Coordinator, TMI _____ Date 8-19-96

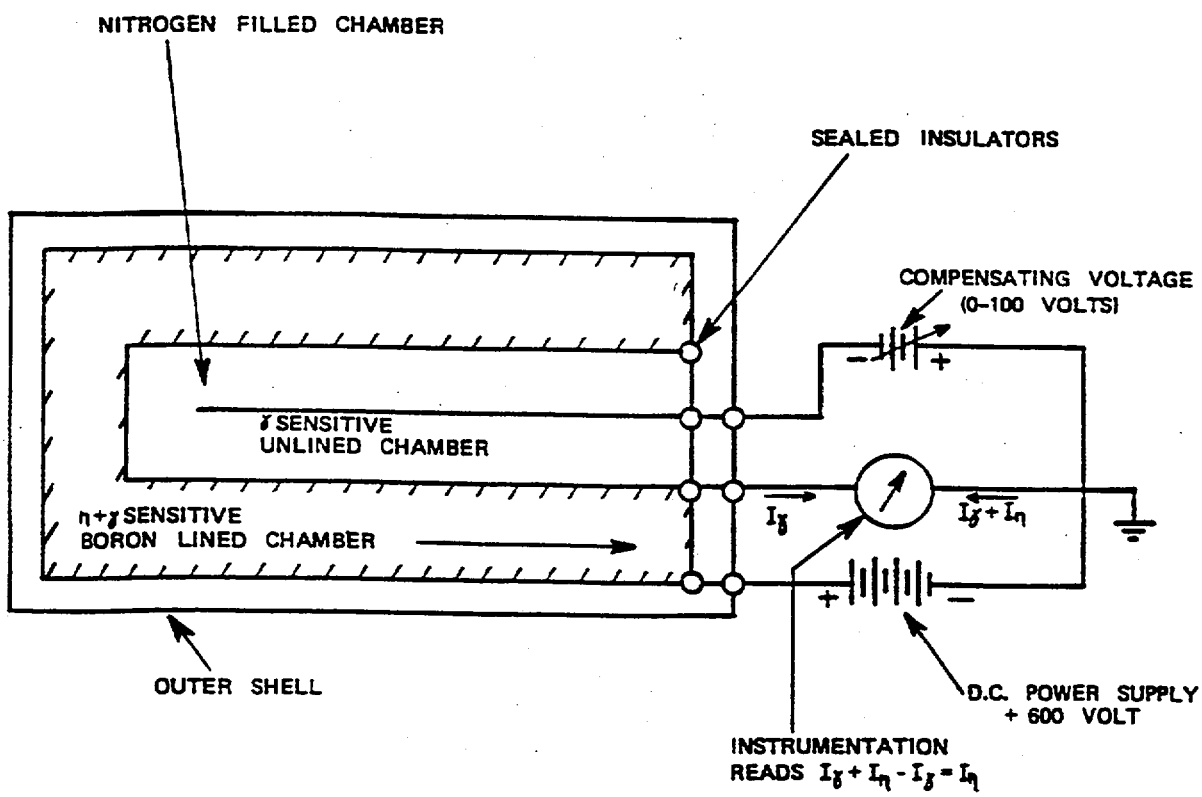
Approved

Plant Operations Director TMI _____ Date 8-26-96

a. Compensated Ion Chamber Operation

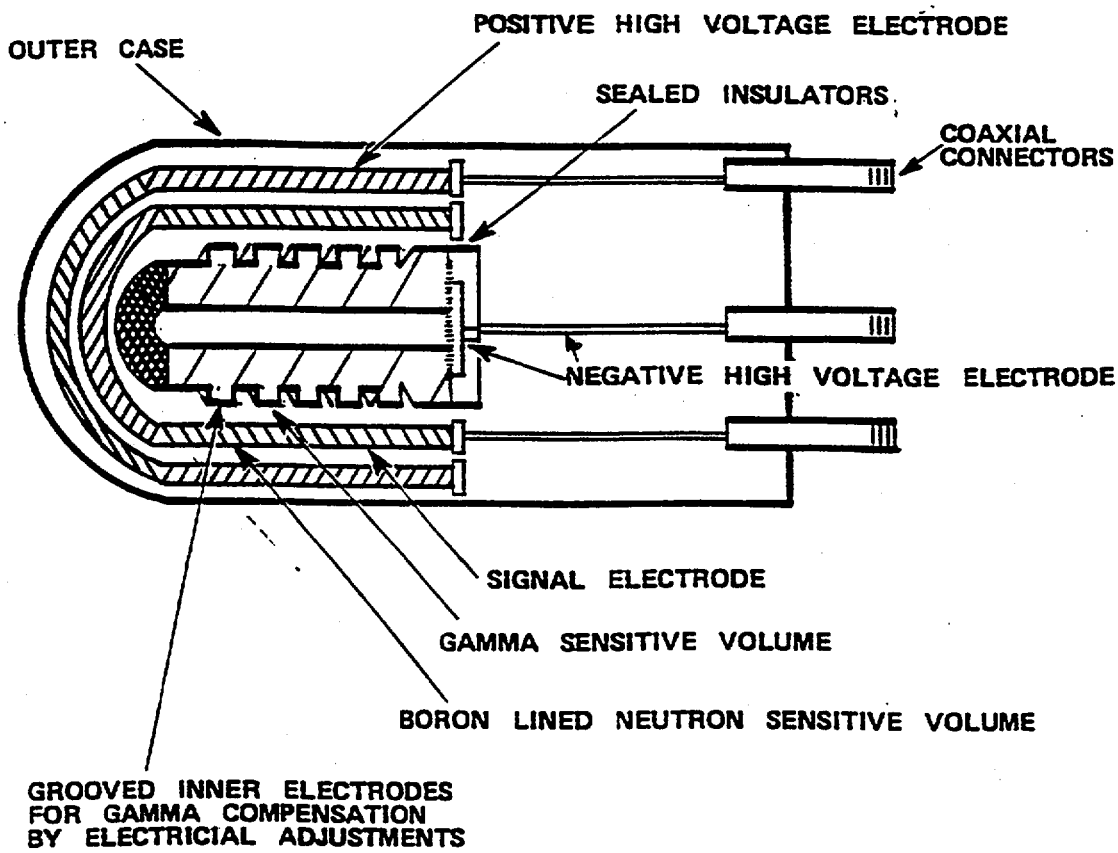
The compensated ion chamber is actually two chambers in one, or two sections in one chamber. One section is sensitive to gammas. The second section is made sensitive to both neutrons and gammas by applying a coating of Boron-10. The outputs of these two chambers are connected in opposition, electrically as shown in Figure 2. Mechanically, the two chambers are generally assembled concentrically as shown in Figure 3. Since the outputs of the two chamber sections are connected in opposition electrically, one of the signals is subtracted from the other. If the output signal coming from one section of the chamber is proportional to a neutron plus gamma signal and the signal coming from the other section of the chamber is proportional to gamma radiation alone, it can be seen from the expression $(\eta + \alpha) - \alpha = \eta$ that when the sections are electrically connected so they subtract from one another, a signal is obtained that is proportional to the neutron radiation alone.

FIGURE 2 - ELECTRICALLY ADJUSTABLE GAMMA COMPENSATED ION CHAMBER



There is one important qualification in the description so far given and that is the assumption that gamma sensitivities can be made equal for both sections of the chamber. Of course, the manufacturer determines this to some extent by selecting the sizes of the two sections of the chamber, but he cannot manage to make it completely constant under all conditions of reactor operating history for all situations of gamma radiation gradients across the chamber. For this reason, the chambers must always be equipped with some scheme for adjusting the constants associated with one of the two sections.

FIGURE 3 - ELECTRICALLY ADJUSTABLE GAMMA COMPENSATED ION CHAMBER



To cope with this problem, The negative electrode which comprises one side of the inner sensitive volume is grooved or serrated. As the voltage is changed on this particular electrode, the electrical field surrounding it tends to change in such fashion as to, if you will, fill up the grooves. This produces an effective change in chamber volume without any mechanical operations. Thus, it is possible to produce a compensating type of adjustment without actually requiring any mechanical adjustment of chamber size. These compensating voltage adjustments are used to correct for variations in manufacturing tolerances and variations in positioning of the chamber adjacent to the reactor.

One way in which to learn more about the problems of chamber compensation is to consider the manner in which improper compensation, either overcompensation or under-compensation, may manifest itself. One common situation in which such evidence is available is when the reactor is shutdown following a period of operation at high power level. If the chamber is reasonably well compensated, the neutron level as displayed on a channel actuated by a compensated ion chamber will show initially the sudden sharp drop in the neutron level changing into the 80 second delayed neutron shutdown period. This shutdown curve should be evident well toward the lower end of the range of the channel in question. If the chamber is under-compensated, one will observe a drop of several decades after which the indication tends to level off and show no further drop. This lack of further drop is an indication of the signal produced by gamma decay of the fission product inventory in the core and indicates that the channel is responding to this more than to neutrons. If the chamber is overcompensated, as the reactor power level drops, it may tend to suddenly drop toward zero, deviating in this fashion from the straight line ordinarily displayed on the logarithmic type instrument. This indicates that the channel is overcompensated and the neutron signal is being suppressed by an erroneously high level of gamma compensation. See Figure 4.

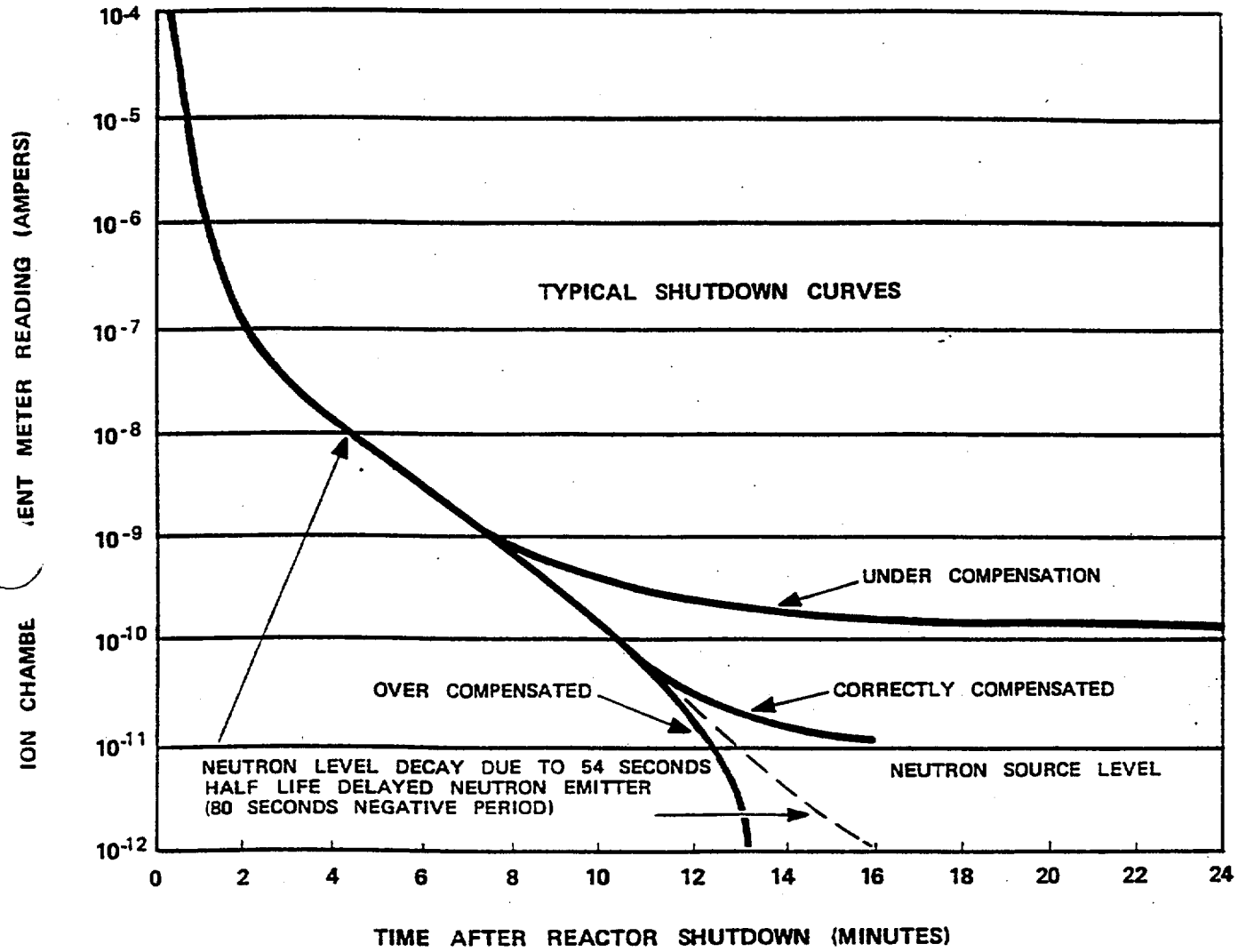
Other possible operational manifestations of chamber over or under compensation can be described. Assume the reactor is operating at full power and the lead supplying the compensating voltage breaks. What effect does the operator see on the indication for that particular channel? If the reactor is operating at a substantial power level, above its normal shutdown condition and the ratio of neutron sensitivity to gamma sensitivity is typical, one would expect to see nothing or very little at any rate.

Another situation can be postulated which is perhaps more informative. Assume that the reactor is operating at some lower power level not far above the normal shutdown level. Also, assume that there is a substantial amount of gamma radiation being released by residual fission products. Assuming these conditions, what happens if the lead to the compensating section of the chamber opens? If the chamber is under-compensated, say radically under-compensated, one will see nothing. There will be no change in the indication. If the chamber is overcompensated, implying that the gamma or residual radiation from the core is producing an excessive chamber output, due to the gamma sensitivity of the chamber, one will tend to see a large increase in the indication coming from this particular chamber. If the chamber is overcompensated, the actual output will tend to be depressed relative to the true neutron level in the core. This being the case, if the compensating voltage disappears, one will see a step increase in indication. Such an increase might have sufficient size and speed. If the chamber is properly and adequately compensated, one will still tend to see a positive increase in indication from the channel. Since the cancellation effect of the compensating channel will disappear, the resulting signal will be due to the neutron signal plus the gamma signal from the one section of the chamber.

Gamma compensation (see Figure 4) is used to overcome the current caused by gamma interaction. Compensated ion chambers are ordinarily used to feed the channels which provide indication of reactor neutron level between the low or source range and the range of power operation.

The need to understand the reactor instrumentation channel which is being fed by a compensated ion chamber is important at all times. However, it is perhaps less of a problem under the conditions when a reactor is being started or restarted from a situation of low residual gamma radiation such as the first startup or after a refueling. Circumstances under which the proper understanding of the operation of the compensated ion channel become most important are those following a sudden shutdown of the reactor. This is a situation that can conceivably be of considerable interest in utility operations since returning a plant to a line following an unscheduled shutdown could be of major importance to the entire system. Under these circumstances, compensation can be a problem. The gamma flux level and the gamma flux spectrum are changing at a rate which may require continual adjustments of compensation. However, it is important to make them correctly.

FIGURE 4 - TYPICAL GAMMA COMPENSATED CURVE FOR A COMPENSATED ION CHAMBER



Form ES-401-6

Written Exam Question Worksheet

Q # 027

Examination Outline Cross-Reference RO SRO

KA # APE 037 AK3.07 Page Tier # 1

RO/SRO Importance Rating 4.2 4.4 Group # 2

Measurement Knowledge of the reasons for the following responses as they apply to the S/G Tube Leak: Actions contained in EOP for S/G tube leak

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

The crew has taken action to minimize subcooling margin in accordance with ATP 1210-5, OTSG Tube Leakage. Identify the ONE (1) statement below that describes the reason for minimizing subcooling margin in this situation.

A. Minimize the differential pressure between the OTSG and RCS.

B. Minimize time required for cooldown of the RCS.

C. Minimize potential of lifting Main Steam safety valves.

D. Minimize tensile stresses on affected OTSG tubes.

Technical Reference OS-24, ATOG Theory and Practice, Rev. 1 page 5

Open Exam Reference

Learning Objective V.E.05.05 Describe the methods available in ATP 1210-5, OTSG Tube Leakage, for limiting integrated OTSG tube leakage.

Question Source ☐ New ☑ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question # 60

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge

☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5, .10 ☐ 55.43 ☑ 55.45 .6, .13

Discriminant Validity Statements

A Correct answer.

B Plausible since depressurization of RCS is required during cooldown.

C Plausible since lowering RCS pressure reduces maximum OTSG pressure if affected OTSG is filled to solid condition.

D Plausible since RCS pressure reduction reduces stress on affected OTSG tubes.

Comments None.

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TMI
Operations Department
Administrative Procedure

Number: OS-24

Title: **ATOG Theory and Practice**

Revision No.: 1

Applicability/Scope: TMI Division

Responsible Office: Plant Operations Dir.

Effective Date: 04/15/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	James R Paules	4/6/99
Procedure Owner	James R Paules	4/6/99
PRG	[illegible]	4/7/99
Approver	James R Paules	4/6/99

4.1.5 Guidelines: Guidelines comprise the second section of procedure steps in ATP 1210-10, Abnormal Transients Rules, Guides and Graphs. They are procedure steps that apply to a number of procedures. The Guidelines have the same force for procedure compliance as any other procedural steps. In general, the Guidelines pertain to equipment initiation and recovery actions. They have an indirect effect on core cooling. Memorization of the Guidelines by the operators is not required. However, the operators are required to know what Guidelines are provided in ATP 1210-10. Identification of specific actions that apply to more than one procedure allows for simplification and standardization of the procedures.

The Guidelines may be implemented without direction from the ATP's if conditions and/or intent of the guideline are met and if the possible consequences of performing the actions in the guideline are evaluated. This flexibility allows performing actions when the operating crew feels that they are needed to enhance transient mitigation. Performance of the guidelines should not be used to replace the verified and validated procedure sequencing and flow. If the procedure would direct performance of the specific guideline of concern within the next few steps and performance of the action(s) is not vital then the procedure sequence should be followed.

4.1.6 Conflicts between Rules & Guidelines: A rule always takes precedence over a Guideline. An example of this condition is the EFW Throttling Criteria Rule takes precedence over the Feeding Dry OTSG Guideline during a loss of SCM.

4.1.7 Applicability of Rules and Guidelines: Rules and Guidelines apply to all modes of plant operations above DHR operating conditions.

4.1.8 RCP Restart: Forced circulation is the preferred mode of operation. Therefore, whenever the RCP Restart Criteria are met at least one RCP should be restarted. If possible, a second RCP should be started in the opposite loop.

4.1.9 OTSG Availability as a Heat Sink: The OTSG's should be maintained available as a heat sink even when primary-to-secondary heat transfer is not required for core cooling. Additional equipment failures, equipment degradation or transient progression might change the requirements for primary-to-secondary heat transfer later in the transient. As a minimum, level and pressure control should be maintained. It is desirable to reduce OTSG saturation temperature to follow the core cooldown. This action will make it relatively easy to make the OTSG a heat sink if and when required. Additionally, tube-to-shell differential temperatures are controlled by steaming the OTSG to follow the cooldown.

4.1.10 Valve Operations: If a valve identified in the ATP's cannot be closed then the corresponding block valve should be closed to perform the isolation function. If a block valve is not included in the system then a valve immediately upstream or downstream should be closed to perform the isolation function.

4.1.11 Definition of Minimize Subcooling Margin: Several conditions require that the operator minimize subcooled margin. This can be achieved by maintaining RCS P-T close to the 25°F SCM limit. The **recommended** band is 30 - 70°F SCM. This band is low enough to meet the requirements of minimizing SCM and wide enough to allow operating margin during a transient. The band provided by the SRO should be based on the goal of minimizing leakage for SGTR events or minimizing pressure stress for PTS concerns.

Form ES-401-6

Written Exam Question Worksheet

Q # 028

Examination Outline Cross-Reference RO SRO

KA # EPE 038 EK3.08 Page 4.1-11 Tier # 1

RO/SRO Importance Rating 4.1 4.2 Group # 2

Measurement Knowledge of the reasons for the following responses as they apply to the SGTR: Criteria for securing RCP

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

The following plant conditions exist:
- Cooling down after a "A" Steam Generator Tube Leak
- RCS pressure 985 psig
- RCS temperature 530°F
- All four Reactor Coolant Pumps operating
- 75°F/hour cooldown rate

Which ONE of the following actions should be taken:

A. Reduce cooldown rate to 50°F/hour or less

B. Increase cooldown rate toward 240°F/hour

C. Secure one Reactor Coolant Pump in each loop

D. Secure all four Reactor Coolant Pumps

Technical Reference ATOG 1210-1, Reactor Trip, Rev. 40 page 4

Open Exam Reference Steam Tables

Learning Objective V.E.01.08 Given the plant conditions determine which IF/THEN step(s) of ATP 1210-1 apply and which other procedure(s) will be used in conjunction with the ATP 1210-1.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge

☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5, .10 ☐ 55.43 ☑ 55.45 .6, .13

Discriminant Validity Statements

A Plausible because this is the cooldown rate limit for natural circulation.

B Plausible because this is the emergency cooldown rate limit.

C Plausible because this is the action after subcooling margin has been minimized.

D Correct answer. Loss of 25°F subcooling requires tripping all four RCPs.

Comments None.

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GPU NUCLEAR	TMI Abnormal Transient Procedure	Number 1210-1
Title **Reactor Trip**		Revision No. 40
Applicability/Scope TMI Division	Responsible Office Plant Operations Dir.	Effective Date 10/19/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator		10/15/99
Procedure Owner	James R Faulks	10/15/99
PRG	John A. DeFork	10/18/99
Approver	James R Faulks	10/15/99

2.7 Verify Safety System Status.

Verify RCS greater than 1600 psig and RB less than 4 psig.

IF an ES actuation setpoint has been reached;	THEN verify all components have been actuated for the respective ES signal(s).

Verify OTSG greater than 600 psig and 10 inches level.

IF an HSPS actuation setpoint has been reached;	THEN verify all components have been actuated for the respective HSPS signal(s).

2.8 Announce Reactor Trip over plant page and radio.

2.9 Verify SCM ≥ to 25°F.

IF SCM < 25°F	THEN trip all RCP's, AND initiate HPI, AND verify EFW, AND raise OTSG level to 75-85%, AND go to ATP 1210-2.

2.10 Verify RCS temperature/pressure and OTSG pressure approaching post trip temperatures and pressures within 2-5 minutes.

IF excessive primary to secondary Heat Transfer	THEN throttle MFW/EFW, AND isolate steam leak, AND increase RCS make-up as necessary AND in the absence of higher priority symptoms go to ATP 1210-3.
IF lack of primary to secondary Heat Transfer	THEN verify MFW/EFW AND in the absence of higher priority symptoms go to ATP 1210-4.

2.11 Verify indications of OTSG tube leakage do NOT exist.

IF an OTSG Tube Leak/ Rupture has occurred;	THEN in the absence of higher priority symptoms go to ATP 1210-5.

Form ES-401-6

Written Exam Question Worksheet

Q # 029

Examination Outline Cross-Reference RO SRO

KA # APE 054 AK3.04 Page 4.2-35 Tier # 1

RO/SRO Importance Rating 4.4 4.6 Group # 2

Measurement Knowledge of the reasons for the following responses as they apply to the Loss of MFW: Actions contained in EOPs for loss of MFW

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

Plant conditions:
- Time is ten minutes after reactor trip due to loss of both Main Feedwater Pumps.
- EF-P-2A is operating, EF-P-1 and EF-P-2B are not operating.
- T-hot is 585° F and slowly increasing.
- RCS pressure is 2300 psig and slowly increasing.
- All RCPs are operating.
- OTSG 1A level is 25 inches.
- OTSG 1B level is 0 inches.
- OTSG 1A pressure is stable at 1010 psig.
- OTSG 1B pressure is 800 psig and decreasing.
- RCS heat up rate is +75° F/Hr.

From the list below, identify the ONE (1) required action concerning operation of the Reactor Coolant Pumps.

A. Stop 1 RCP per loop.

B. Stop 3 RCPs.

C. Stop 4 RCPs.

D. Continue to operate 4 RCPs.

Technical Reference ATOG ATP 1210-4, Lack of Primary to Secondary Heat Transfer, Rev. 21, Page 2. Steam Tables.

Open Exam Reference Steam Tables.

Learning Objective V.E.04.07 Given plant conditions determine if entry is required into ATP 1210-4, Lack of Heat Transfer, and state the required immediate actions.

Question Source ☐ New ☑ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question # 19

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge

☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5, .10 ☐ 55.43 ☑ 55.45 .6, .13

Discriminant Validity Statements

A Correct answer.

B Plausible distracter since this reduces heat input (by the RCPs) into the RCS during conditions of inadequate heat transfer while still maintaining forced flow through the reactor core which is the preferred method of core cooling.

C Plausible distracter since this maximizes the reduction of heat input (by the RCPs) into the RCS during conditions of inadequate heat transfer.

D Plausible distracter since this maintains maximum possible flow through the reactor core during conditions of inadequate heat transfer.

Comments None.

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TMI
Abnormal Transient Procedure

Number: 1210-4

Title: **Lack of Primary to Secondary Heat Transfer**

Revision No.: 21

Applicability/Scope: TMI Division

Responsible Office: Plant Operations Dir.

Effective Date: 10/15/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	[illegible]	9-22-99
Procedure Owner	James N Paules	9/24/99
PRG	[illegible]	9/27/99
Approver	James N Paules	9/24/99

1.0 **IMMEDIATE ACTIONS**

1.1 Reduce RCP's to one per loop.

1.2 IF RCS pressure increases to the PORV setpoint **WHILE ATTEMPTING TO RESTORE FEEDWATER**; THEN go to ATP 1210-9 to initiate HPI cooling.

1.3 Initiate FW.

1.3.1 Start EF-P1. EF-P2A, and EF-P2B. Open EF-V-30A or D and B or C.

1.3.2 IF EFW is NOT available; THEN use Main FW.

1.3.3 IF neither MFW nor EFW are available; THEN attempt to restore emergency feedwater, refer to ATP 1210-10, Guideline 2.10.

1.4 IF neither MFW nor EFW can be established; THEN go to ATP 1210-9 to initiate HPI cooling.

2.0 **FOLLOWUP ACTIONS**

_____ 2.1 If the EAL listed below has been met, implement EPIP-TMI-.01, EMERGENCY CLASSIFICATION, and additional implementing procedures as required.

SITE AREA EMERGENCY
S4.4 (HSD Function) 1) Loss of all means to feed AND steam OTSGs **AND** 2) Loss of RCS makeup and PZZR level < 20" APPLICABILITY: Pwr Ops, HStby, SU, HSD

_____ 2.2 Control OTSG level based on RCP operation and subcooled margin.

_____ 2.3 IF there is a dry OTSG; THEN minimize SCM and refer to ATP 1210-10, Guideline 2.11.

☐ 2.4 WHEN primary to secondary heat transfer has been established; THEN go to Step 2.14.

☐ 2.5 IF RCS pressure increases to PORV setpoint **WHILE ATTEMPTING TO ESTABLISH PRIMARY TO SECONDARY HEAT TRANSFER**; THEN open the PORV and PORV block UNTIL RCS pressure decreases to 100 psig above highest OTSG pressure OR RCS pressure approaches 25°F SCM.

☐ 2.6 WHEN **RCP RESTART CRITERIA** are met; THEN restart one RCP per loop, refer to ATP 1210-10, Guideline 2.14.

Form ES-401-6

Written Exam Question Worksheet

Q # 030

Examination Outline Cross-Reference RO SRO

KA # APE 058 AK1.01 Page 4.2-43 Tier # 1

RO/SRO Importance Rating 2.8 3.1 Group # 2

Measurement Knowledge of the operational implications of the following concepts as they apply to the Loss of DC: Battery charger equipment/instrumentation

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** A

The following alarms, among others, are received simultaneously:
- A-1-7 Battery 1A Discharging
- A-2-7 Batt Charger 1A/1C/1E Trouble
- A-3-7 Inverter 1A/1C/1E System Trouble
- PRF1-1-1 CRDM Breaker Test Trouble
- NN-3-1 230 KV Substation Trouble
- AA-3-2 7KV Bus Trouble
- AA-3-3 4KV BOP Bus Trouble
- AA-3-5 480V BOP Bus Trouble

Which ONE (1) of the following events would result in ALL of the above indications?

A. Loss of "A" DC Distribution

B. Loss of the A/C/E Inverters

C. Loss of "A" Battery Charger

D. Loss of "A" Aux Transformer

Technical Reference EP 1202-9A, Loss of "A" DC Distribution System, Rev. 41 page 2

Open Exam Reference

Learning Objective V.D.04.04 Given a list of symptoms, identify the event, determine what Immediate Automatic Actions will occur and what Immediate Manual Actions are required IAW EP 1202-9A.

Question Source ☐ New ☑ TMI Bank TMI Question # QR5D04-04-Q01

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge

☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .8, .10 ☐ 55.43 ☑ 55.45 .3

Discriminant Validity Statements

A Correct answer.

B Plausible due to inverter alarms.

C Plausible due battery discharging alarm.

D Plausible due to substation and bus trouble alarms.

Comments None.

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GPU NUCLEAR	TMI Emergency Procedure	Number 1202-9A
Title: **Loss of "A" DC Distribution System**		Revision No. 41
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: 08/19/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

TRAINING DEPT.

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	Signature	Date
Originator	Brantley for Doug Gorse	8-6-99
Procedure Owner	Brantley	8-6-99
PRG	[illegible]	8/12/99
Approver	James Vanlos	8/10/99

NOTE

Partial loss of DC caused by a blown fuse to one of the DC Distribution Panels may give many of the same symptoms, however, this procedure does not provide specific guidance for a blown fuse.

1.0 SYMPTOMS

A. Loss of Main Distribution Panel 1A as indicated by alarms:

1. AA-3-2, 7KV Bus Trouble
2. AA-3-3, 4KV BOP Bus Trouble
3. AA-3-5, 480V BOP Bus Trouble
4. A-1-7, Battery 1A Discharging (Rate above 100 amps)
5. A-2-7, Battery Charger 1A/1C/1E Trouble
6. A-3-7, Inverter 1A/1C/1E Trouble
7. K-3-4, MN Turb. PC oil pmp strt/troub
8. L-1-3, Voltage Regulator DC Loss
9. B-3-1, 4KV ES Bus Trouble
10. NN-3-1, 230KV Substation Trouble (loss of DCA)
11. PRF-1-1-1, CRDM Breaker Test Trouble (loss of shunt trip)
12. H & V, A-4-2 Cont. Bldg. Batt. Chargers A Damper Tbl, Fire-Smoke
13. Loss of breaker status lights at control switches

2.0 IMMEDIATE ACTION

A. Automatic Action:

1. Loss of Main Distribution Panel 1A will result in loss of all power on the "A" Distribution System, resulting in the inability to remotely trip or close breakers on A ESAS System.
2. Loss of Engineered Safeguards Distribution Panel 1E
3. Loss of ES Diesel Generator Dist. Pnl. 1P
4. Loss of 230KV Substation Dist. Pnl. DCA
5. Loss of Distribution Panel 1C
6. Loss of Distribution Panel 1H

Form ES-401-6

Written Exam Question Worksheet

Q # 031

Examination Outline Cross-Reference RO SRO

KA # APE 059 AK2.02 Page Tier # 1

RO/SRO Importance Rating 2.7 2.7 Group # 2

Measurement Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following: Radioactive-gas monitors

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

The RCDT is being pumped down when RM-G-20, (RC Drain Pump Discharge Monitor) High Alarm actuates.

From the list below, identify which ONE (1) of the following automatic actions will occur as a result of this event.

A. WDL-V-534 AND WD-L-535 CLOSE

B. WDG-V-3 AND WDG-V-4 CLOSE

C. CA-V-4B AND CA-V-5B CLOSE

D. CA-V-2 AND CA-V-13 CLOSE

Technical Reference EP 1202-12, Excessive Radiation Levels, Rev. 45, Page 13.

Open Exam Reference

Learning Objective IV.E.06.04 Given an RMS channel, state the interlocks associated with a channel in the Radiation Monitoring system.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .7

Discriminant Validity Statements

A Plausible since this isolates the RB sump.

B Correct answer.

C Plausible since this isolates OTSG "B" sample lines.

D Plausible since this isolates the RCS Sample line.

Comments None.

CONTROLLED COPY

GPU NUCLEAR	TMI Emergency Procedure	Number 1202-12
Title: **Excessive Radiation Levels**		Revision No. 45
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: 10/11/99

This document is within QA plan scope: [X] Yes [ ] No

Safety Reviews Required: [X] Yes [ ] No

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	Signature	Date
Originator	[illegible]	10/2/99
Procedure Owner	[illegible]	10/2/99
PRG	[illegible]	10/5/99
Approver	James R Paulos	10/3/99

ENCLOSURE 1

2.0 Interlock functions actuated by Hi Alarms from specific liquid radiation monitor are as follows:

_____ 2.1 RCS Letdown Line Monitor, RM-L-1 Hi.

RCS Letdown Isolation Valves MU-V-2A and MU-V-2B closes.

_____ 2.2 Radioactive Waste Water Discharge Monitor, RM-L6.

Mechanical Draft Cooling Tower Stop Valve, WDL-V257, closes.

_____ 2.3 Plant Discharge Line Monitor, RM-L7.

Mechanical Draft Cooling Tower Stop Valve, WDL-V257, closes.

_____ 2.4 IWTS/FS Discharge Line Monitor, RM-L12

The following pumps/valves trip off/shut.

a. IW-P-16, 17, 18, IW-V-73 - Effluent pumps/valve in IWTS.

b. IW-P-29, 30, IW-V-279 - Effluent pumps/valve in IWFS.

3.0 Interlock Functions actuated by Hi Alarms from specific area monitors

_____ 3.1 "A" O.T.S.G. Sample Line Monitor RM-G-16

"A" O.T.S.G. Sample Line Isolation Valves CA-V-4A and CA-V-5A close.

_____ 3.2 "B" O.T.S.G. Sample Line Monitor RM-G-17

"B" O.T.S.G. Sample Line Isolation Valves CA-V-4B and CA-V-5B close.

_____ 3.3 R.C.S. Sample Line Monitor RM-G-18

R.C.S. Sample Line Isolation Valves CA-V-1, CA-V-2, CA-V-3 and CA-V-13 close.

_____ 3.4 R.C. Drain Pump Discharge Line Monitor RM-G-20

a. R.C. Drain Pump Discharge Isolation Valves WDL-V-304 and WDL-V-303 close.

b. R.C. Drain Tank Vent Isolation Valves WDG-V-3 and WDG-V-4 close.

_____ 3.5 Reactor Building Sump Drain Monitor RM-G-21

Reactor Building Sump Drain Line Isolation Valves WDL-V-534 and WDL-V-535 close.

Form ES-401-6

Written Exam Question Worksheet

Q # 032

Examination Outline Cross-Reference RO SRO

KA # APE 060 AA2.06 Page 4.2-47 Tier # 1

RO/SRO Importance Rating 3.6 3.8 Group # 2

Measurement Ability to determine/interpret the following as they apply to the Accidental Gaseous Release: Valve lineup for release of radioactive gases

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

During a waste gas release, which ONE (1) of the following provides protection against an accidental gas release?

A. The inlet valves are interlocked to prevent adding additional gas to a tank that is releasing.

B. The gas release will terminate if the tank outlet flow monitor falls below setpoint.

C. The outlet valves are interlocked to prevent opening more than one at a time.

D. The gas release will terminate if the Aux Building supply fan trips.

Technical Reference OPM E-04, Watse Disposal Gaseous System, page 9, rev 8.

Open Exam Reference

Learning Objective IV.B.08.08 Describe the flow path for a normal gas release and explain the following actions concerning the gas release.
a. initiating
b. monitoring
c. securing
d. interlocks

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5 ☑ 55.45 .13

Discriminant Validity Statements

A Correct answer.

B Plausible since this instrument provides isolation on high flow.

C Plausible since these valves are administratively controlled.

D Plausible since Aux Bldng exhaust fan is required for gas release for dilution flow and examinee may expect a supply fan is required also.

Comments None.

FOR INFORMATION ONLY

SECTION E-04
REVISION 8

TMI OPERATIONS PLANT MANUAL
WASTE DISPOSAL GASEOUS SYSTEM

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Concur

Owner James R Powles Date 12/12/96

Operations Training Manager, TMI [illegible] Date 12/16/96

Training Coordinator, TMI Linda L. Ritter Date 12-17-96

Approved

Plant Operations Director TMI L. G. Nelle Date 12-18-96

7.5.4 The controls integral with each unit automatically makeup or drain water, as required, from the de-entrainer tank. A discharge control valve from the de-entrainer tank ensures that the pressure therein is always maintained at 50 psig regardless of how low the pressure is in the Waste Gas Decay Tank to which the compressor may be discharging. Another control valve, in the recycle discharge line from the de-entrainer tank back to the suction of the compressor, opens as required to always maintain a minimum pressure at the compressor suction (1.2 psig for TMI Unit 1) and closes proportionately to the positive differential between actual suction pressure and the minimum (the recycle gas control valve is fully closed when the inlet pressure to the compressor is 1.7 psig). Another control valve adds nitrogen gas to the recycling compressor, via the recycle gas line, as required to maintain a minimum of 50 psig in the de-entrainer tank when the compressor is recycling gas.

7.6 Modes of Operation

7.6.1

PSIG	PSIA	VALVE	COMPRESSOR
1.6	16.3	WDG-V-18 OPENS	WDG-P-1A STARTS
1.8	16.5	WDG-V-21 OPENS	WDG-P-1B STARTS
1.4	16.1	WDG-V-21 CLOSES	WDG-P-1B STOPS
1.2	15.9	WDG-V-18 CLOSES	WDG-P-1A STOPS

7.6.2 Manual - Suction valve will open and compressor will start and run until manually stopped.

7.6.3 Abnormal - If one compressor is started in manual and trips because of overload or breaker is opened the other compressor will auto-start at setpoint but will not stop automatically. It must be stopped manually.

7.7 Decay Tank Sequencer

7.7.1 Uses WDG-PS-420/421/422 (set at 80#) and position limit switches on WDG-V-24 thru 32.

7.7.2 At 80# in tank being filled it searches for another tank for fill. A-B-C-A rotation.

7.7.3 Fill Acceptance Interlock

a. New tank must be < 80 psig.
b. BOTH outlet valves closed.

Form ES-401-6

Written Exam Question Worksheet

Q # 033

Examination Outline Cross-Reference RO SRO

KA # APE 061 AA2.04 Page 4.2-48 Tier # 1

RO/SRO Importance Rating 3.1 3.5 Group # 2

Measurement Ability to determine and interpret the following as they apply to the Area RM System Alarms: Whether an alarm channel is functioning properly

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

Which ONE (1) of the following is NOT a method of evaluating whether a Radiation Monitoring System channel is operating properly?

A. Comparing the meter reading with the recorder trace.

B. Comparing the meter reading with the Plant Process Computer point.

C. The meter indication is on-scale without the check source.

D. Obtaining a meter reading increase when performing a source check.

Technical Reference SP 1301-1, Shift and Daily Checks, Rev. 136 pages 7, 8 & 20

Open Exam Reference

Learning Objective IV.E.06.06 Explain how to source/cal check various radiation monitors.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5 ☑ 55.45 .13

Discriminant Validity Statements

A Plausible, the examinee may not consider the recorder part of the channel's functionality.

B Plausible, the examinee may not consider the PPC point part of the channel's functionality.

C Correct answer. Does not include a qualitative assessment with known plant conditions.

D Plausible, the examinee may not consider this a good qualitative assessment.

Comments None.

CONTROLLED COPY

AmerGen	TMI - Unit 1 Surveillance Procedure	Number 1301-1
		Revision No. 136
Shift and Daily Checks Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date 03/28/00

This document is within QA plan scope [X] Yes [ ] No
Safety Reviews Required [X] Yes [ ] No

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8.2.2 If the plant computer is unavailable, complete Enclosure 2 and note the following steps.

8.2.2.1 Obtain I&C assistance and place RPS channel in manual bypass to obtain voltage measurements in the RPS cabinets.

8.2.2.2 Use RPS RCS pressure instruments and RPS T hot indications to calculate A & B loop Tsat for comparison to RC-TI-977 and 978.

8.2.3 At console PRF (or RM-L12 location), depress C.S. (Check Source) button for the channel under test. Look for a definite increase in CPM. Circle "Y" if a definite increase was observed.

8.2.4 Rod index is provided by the plant computer, use NI power (NI-5 or equivalent) for Rx power to determine rod index limit.

8.2.5 MU Tank Pressure indication in the Control Room is compared to the adjusted suction pressure of a MU pump which is not operating in order to check the operability of the MUT pressure channel. The 3 psi subtracted from the pump suction pressure indication is to correct for the elevation difference and head loss due to flow between the measured suction pressure and the MU Tank gas pressure.

8.3 Data Sheet 3

8.3.1 OTSG leak rate determination is described in Enclosure 1.

8.3.2 Prior to performing SP 1302-1.1, review NAS group 2 to verify that the heat balance inputs are valid.

8.4 Data Sheet 4

8.4.1 Verify the readings for liquid and gas effluent flow are within the normal expected range for current plant conditions.

8.4.2 Daily Fire Door Inspections are required by SP 1303-12.21, SP 1303-12.20 and SP 1303-12.22. These inspections are normally performed by an Auxiliary Operator and returned to the Control Room for review by a licensed operator. Initialing the 1301-1 Data Sheet documents that the schedule and surveillance content are satisfied or remedial action initiated. There is no need to attach the complete fire door surveillance to this document.

8.4.3 Radiation Monitor Sensor and Recorder Check (Daily)

8.4.3.1 At Console PRF (or ALC-RMI-18 location), read the background indication for each applicable RMS channel. Record the values on Data Sheet 4.

NOTE

If the indicated background level is greater than twice the radiation level which the check source should exhibit, an increase due to check source actuation may not occur. In this case mark "N/A" in the "Sensor Check" space on the data sheet. If no increase was observed on RM-L6, perform Step 8.4.3.2.

8.4.3.2 Special source check for RM-L6

a. For RM-L6, remove the lead plug from the source well in the sampler and insert Met-Ed Rod Source No. 192 (11.37 μCi Cs^{137}). Record resulting CPM on the data sheet. Compare this reading with the background recorded in Step 8.4.3.1. A minimum increase of 1×10^5 CPM is required. Remove the source and reinsert the lead plug.

8.4.3.3 Compare the panel meter and recorder indication (where available) for each channel being checked. Look for a significant difference which would imply miscalibration or malfunction.

8.4.3.4 Compare the panel meter and PPC indication for each channel being checked. Look for a significant difference which would imply miscalibration or malfunction.

8.4.3.5 RM-A14 ESF Ventilation Radiation Monitor Shiftly Checks

1. Test to be performed in Control Room at Remote Digital Readout Panel (RDR).

NOTE

At start and end of checks digital readout should be selected for μCi/cc.

2. Perform Check Source Tests in the following order.

A. Normal Range Detector

1) Place Program Mod Keyswitch to "Enter Data".

2) Depress "DISP OP" button.

3) Depress "1" button.

4) Depress "ENTER" button.

5) Place Program Mod Keyswitch to "RUN" and record background CPM.

6) Depress CS button until light comes on.

DATA SHEET 4

Daily Checks (Night Shift)

Sections A, C, D, E, G, H, I, and K to be completed at 2100 ± 2 hours

8.4A ODCM Part I, 3.1.2 and 3.1.1 **(to be completed at 2100 ± 2 hours).**

When required, perform source check and verify that recorder & meter indications are comparable.

Required	Channel	Record (CPM) Background	Expected Increase Above Bkgrd	Source Check OK? Circle Y or N	PPC/Recorder Pt. vs Meter OK? Check √ (OK) PPC Pt./Recorder	
At all times	RM-A4P		400	Y/N		
	RM-A4I		80	Y/N		
	RM-A4G		150	Y/N		
At all times	RM-A6P		200	Y/N		
	RM-A6I		100	Y/N		
	RM-A6G		200	Y/N		
At all times	RM-A8P		30	Y/N		
	RM-A8I		40	Y/N		
	RM-A8G		70	Y/N		
During RB purge (NIS if RB purge is not in progress)	RM-A9P		40	Y/N/NIS		
	RM-A9I		17	Y/N/NIS		
	RM-A9G		180	Y/N/NIS		
If condenser vacuum is established	RM-A5		147	Y/N		
	RM-A15		104	Y/N		
During ESF vent operation (NIS if EST vent is not operating)	RM-A14G		>800	Y/N/NIS		
	RM-A14G Hi		>4.80 x 10^{-9} amps	Y/N/NIS		
At all times	RM-A7					
	RM-L6		1450	Y/N		
Not T.S. required	RM-L7		1400	Y/N		
During CCB vent operation (NIS if CCB vent is not operating)	ALC-RMI-18P		200	Y/N/NIS		
	ALC-RMI-18I		200	Y/N/NIS		
	ALC-RMI-18G		200	Y/N/NIS		

NOTE

Attach printout of Area 21, Group 13 or Area 19, Groups 1, 3, 4, 5.

If required (see Sect. 8.4.3.2) was RM-L6 source check per Sect. 8.4.3.2.a acceptable? (Circle Y/N) Y/N/Not Required

Performed By_____ Time:_____

Form ES-401-6

Written Exam Question Worksheet

Q # 034

Examination Outline Cross-Reference RO SRO

KA # APE 028 AA2.10 Page 4.2-23 Tier # 1

RO/SRO Importance Rating 3.3 3.4 Group # 3

Measurement Ability to determine or interpret the following as they apply to the Pressurizer Level Control Malfunctions: Whether the automatic mode for PZR level control is functioning improperly, necessity of shift to manual modes

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

The plant is at 100% power with ICS in full automatic operation.

The following plant conditions exist:
- Pressurizer level 200" and decreasing about 4" per minute
- Makeup Tank level 80" and decreasing about 2" per minute
- Makeup flow 60 gpm and increasing slowly
- MU-V-17 demand at 50% and increasing slowly

Which ONE of the following failures would explain the conditions above?

A. MU-V-5 is failing open

B. MU-V-17 is failing open

C. Seal Injection flow instrument is failing low

D. Pressurizer level instrument is failing low

Technical Reference EP 1202-29, Pressurizer System Failure, Rev. 54 page 11

Open Exam Reference

Learning Objective 0008140401 When pressurizer level control is in automatic, respond to a malfunction in pressurizer level indication or control IAW, EP 1202-29, Pressurizer System Failures.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .5 ☑ 55.45 .13

Discriminant Validity Statements

A Plausible because initially Pressurizer level would decrease, MU flow would increase and MU-V-17 would open, but MU Tank level would not decrease.

B Plausible because MU flow would increase and MU Tank level would decrease, but MU-V-17 demand would not increase and Pressurizer level would not decrease.

C Plausible because initially MU tank level will decrease due to the added makeup through seal injection, but Pressurizer level would initially increase causing MU-V-17 demand and MU flow to decrease.

D Correct answer. With MU Tank level decreasing, MU flow and MU-V-17 demand increasing, Pressurizer level should be increasing. Therefore in the absence of an RCS leak, the Pressurizer level instrument must be failing low.

Comments None.

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GPU NUCLEAR

TMI
Emergency Procedure

Number: 1202-29

Title: **Pressurizer System Failure**

Revision No.: 54

Applicability/Scope: TMI Division

Responsible Office: Plant Operations Dir.

Effective Date: 07/06/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator		6-21-99
Procedure Owner		6-24-99
PRG		6/29/99
Approver	James R Faulks	6/25/99

SECTION D

Malfunction In Pressurizer Level Indication Or Control

D.1 Symptoms

1. Disagreement between pressurizer level indicators (computer and console) of more than 12 inches. RC1-LT1 (C1720), RC1-LT3 (C1722) and RC-LI-777.

2. Rapid change in indicated/recorded level due to loss of compensation or loss of power or d/p cell failure or other malfunction, of the pressurizer.

3. Possible high or low pressurizer level alarms.

   a. G-1-5, Pzr Level Hi-Hi

   b. G-2-5, Pzr Level Hi/Lo

   c. G-3-5, Pzr Level Lo-Lo

4. Pressurizer level indicator(s) not responding to changes in pressurizer level.

5. Hi makeup flow alarm (D-3-1, MU Flow Hi).

6. Pressurizer temperature fails to agree with saturation temperature for RCS pressure.

7. RCS pressure changes don't agree with PZR level changes.

D.2 Immediate Action

A. Automatic Action

1. If indication fails low

   a. Pressurizer heaters trip at 80 inches.

   b. Makeup valve MU-V-17 opens.

2. If indication fails high

   a. Makeup valve MU-V-17 closes.

B. Manual Action

1. Take MU-V-17 under hand control and adjust makeup flow to equal letdown flow minus seal injection to maintain makeup tank as constant as possible.

2. Select alternate pressurizer level transmitter.

3. Select alternate pressurizer temperature transmitter.

Form ES-401-6

Written Exam Question Worksheet

Q # 035

Examination Outline Cross-Reference RO SRO

KA # BW A08 2.4.4 Page 2-11 Tier # 1

RO/SRO Importance Rating 4.0 4.3 Group # 3

Measurement Refueling Canal Level Decrease: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency/abnormal procedures

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

The following plant conditions exist:
- The plant is shutdown for refueling.
- Fuel transfer operations are in progress from the SFP to the Reactor Building
- Spent fuel cooling is in operation.

Which ONE (1) of the following would require entry into AP 1202-43, Transfer Canal Seal Plate Gasket Failure/Level Loss?

A. Observed leakage from primary shield penetrations.

B. A fuel assembly is dropped and dents the bottom of the Spent Fuel Pool.

C. Nuclear Service Closed Cooling surge tank is rising at 1 inch per minute.

D. RB pressure less than fuel handling building pressure when opening FH-V-1A/B.

Technical Reference AbP 1203-43, Transfer Canal Seal Plate Gasket Failure/Level Loss, page 2, Rev. 6.

Open Exam Reference

Learning Objective V.C.16.01 Given a list of symptoms, identify the event as a failure of the fuel transfer canal seal plate per AP 1203-43.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .10 ☑ 55.43 .2 ☑ 55.45 .6

Discriminant Validity Statements

A Correct answer.

B Plausible due to damage to the pool liner, but no leakeage occurs.

C Plausible if there was leak in the SFP heat exchanger, but NSCC operates at a higher pressure.

D Plausible if the DP was in the opposite direction.

Comments None.

FOR INFORMATION ONLY

GPU NUCLEAR	TMI Abnormal Procedure	Number 1203-43

Title: **Transfer Canal Seal Plate Gasket Failure/Level Loss**

Revision No.: **6**

Applicability/Scope	Responsible Office	Effective Date
TMI Division	Plant Operations Dir.	12/15/98

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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Originator	[signature]	12/8/98
Procedure Owner	L.G. Nolle for T.H. Goodlander	12-10-98
PRG	[signature]	12/10/98
Approver	L.G. Nolle	12-10-98

1.0 **SYMPTOMS**

1. Rapid decrease in Fuel Transfer Canal Water Level.
2. R.B. Sump Level increasing faster than normal.
3. Fuel Transfer Canal Lo Level Alarm on PLB-4-9.
4. Possible higher than normal radiation levels as indicated on RM-G6, RM-G7, RM-G22, RM-G23, and/or RM-G9.
5. Leakage from primary shield penetrations.

2.0 **IMMEDIATE ACTION**

A. Automatic Action

None

B. Manual Action

> **NOTE**
>
> In some instances time required to respond could be critical depending on size of leak.

1. Evacuate all unnecessary personnel from Reactor Building by sounding the Reactor Building Evacuation Alarm and making an announcement on the page system.
2. If fuel transfer canal level decrease rate is >6" per minute, perform the following steps as quickly as possible:
   a. If any fuel bridge has a component engaged, proceed to the nearest available storage space and lower the fuel component, Do Not Disengage.
   b. If the bridge is closer to the core, insert the component to the full down position in the nearest available core location.
   c. If the bridge is nearer the deep end of the transfer canal, lower the component between the plenum and up ender until the component just touches the fuel pool floor, as noticed by a rapid decrease in load cell reading. Do not disengage the component.
   d. Move the fuel transfer carriages to the spent fuel pool and close FH-V-1A and FH-V-1B.

Form ES-401-6

Written Exam Question Worksheet

Q # 036

Examination Outline Cross-Reference RO SRO

KA # APE 056 AK3.01 Page 4.2-37 Tier # 1

RO/SRO Importance Rating 3.5 3.9 Group # 3

Measurement Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Order and time to initiation of power for the load sequencer

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** D

The plant has experienced a simultaneous Loss of Offsite Power and ESAS actuation.

Which ONE (1) statement below identifies the start times for the affected loads AND the reason for block loading?

A. MU-P-1A starts immediately; DC-P-1A starts 10 seconds later to ensure large loads are energized first.

B. DH-P-1A starts immediately; RR-P-1A starts 10 seconds later to ensure large loads are energized first.

C. MU-P-1A starts immediately; RR-P-1A starts 10 seconds later to prevent overloading the EDG.

D. DH-P-1A starts immediately; DC-P-1A starts 10 seconds later to prevent overloading the EDG.

Technical Reference OPM F-06, Engineered Safeguard Actuation System, pages 8, 10, 11, and 16, Rev 10.

Open Exam Reference

Learning Objective IV.G.06.08 Describe "block loading" and "load shedding" and explain when each occurs for the class 1E electrical system.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5, .10 ☐ 55.43 ☑ 55.45 .6, .13

Discriminant Validity Statements

A Plausible if misunderstanding exists on the basis for block loading.

B Plausible if misunderstanding exists on the basis for block loading and equipment in each block..

C Plausible if misunderstanding exists on equipment in each block.

D Correct answer.

Comments None.

FOR INFORMATION ONLY

SECTION F-06
REVISION 10

TMI OPERATIONS PLANT MANUAL

ENGINEERED SAFEGUARD ACTUATION SYSTEM

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Concur

Owner _____ Date 11/25/99

Operations Training Manager, TMI _____ Date 12/1/99

Training Coordinator, TMI _____ Date 12-2-99

Approved

Plant Operations Director TMI _____ Date 12/3/99

5.1.5 Actuation Cabinets

Output signals of all three relay cabinets for one actuation system are fed to the system's actuation cabinet (i.e., output relay contacts in the sections A of the relay cabinets are wired to actuation cabinet A). A total of two actuation cabinets are provided. The actuation cabinets contain the output relays that actuate the ESAS equipment. Each cabinet is divided into four separate compartments, 4A through D ("A" actuation) and 5A through D ("B" actuation). Compartments 4A through C and 5A through C contain the output relays of the channels selected to make the two-out-of-three logic matrixes for each component to be actuated. Compartments 4D and 5D contain the manual actuation auxiliary relays, which back up the automatic actuation and operate the same components. Manual actuation relays are energized to actuate.

All automatic ES relays are de-energized to actuate. Each auto-actuation cabinet contains a set of load sequence timers which actuate various ESAS components at 5, 10 and 15 seconds after the initiation signal. In the event the Emergency Diesel Generators are required to supply power, the block sequencing provides even loading of them to full load; and, under normal power conditions, they limit the voltage dip on the 4160V bus.

5.1.6 Status Indication

All ESAS system analog signals are indicated at the system channel bistable cabinets and are monitored by the unit computer. Refer to Figure 1 B/S cabinets for these analog signals. The status of each relay is monitored by red neon indicating lights on the relay cabinets and actuation cabinets. The emergency status of each group of output relays is indicated on the control room panel PCR. The status of all the components required during an ESAS actuation are indicated on control room panel PCR; blue and amber lights are used respectively to indicate when a component <u>is</u> or <u>is not</u> in its safeguards position. To ensure positive status indication, the bulbs are fed from either vital bus A or B with the possibility of manual transfer from one to the other at panel PCR.

5.2 Automatic Initiation

5.2.1 Signal Sequence

The anticipated signal sequence for a major loss of coolant accident (LOCA) would be in the following order:

a. Reactor trip at 1900# RCS pressure (reactor trip isolation)

b. High Pressure Injection at 1600# RCS pressure

c. Reactor Building Isolation and Cooling at 4# Building Pressure

d. Partial containment isolation and building spray actuation at 30# building pressure

e. Low Pressure Injection at 500# RCS pressure.

If, however, HPI is not bypassed, and RCS pressure drops below 1600# the 1600# trip bistable switches to contact #1 closed and #2 open. This deenergizes the T3/- relay, opening the T3/- contact which deenergizes the 63Y1 relay, opening the 63Y1 contact causing the actuation relays 63Z1 and 63Z2 to deenergize actuating block 1 loads. Another 63Y1 contact opens in the block timing relay circuit actuating block 2, 3 and 4 loads in the following manner:

Block 1

a. Diesels start but do not load unless there is an undervoltage condition on the 1D or 1E 4160 Volt ES Buses.

b. ES selected makeup pumps start

c. MU-V-14A and 14B open

d. MU-V-16A, 16B, 16C and 16D open

e. MU-V-36, 37 shut

f. NR-V-4A and 4B shut

g. AH-E-1A, 1B and 1C stop

h. DH-P-1A and 1B start

i. DH-V-4A and 4B open

j. DH-V-5A and 5B open (normally open)

k. Lo HPI flow alarm activated (75 gpm)

l. MU-V-2A, 2B and 18 shut

m. CM-V-1, 2, 3 and 4 shut

n. RB-V-2A and 7 shut

o. RR-V-1A and 1B open

p. RR-V-3A, 3B, 3C open (normally open)

q. RR-V-4A, 4B, 4C and 4D open

r. MU-V-3, closes

5 seconds after block 1 initiates, block 2 initiates.

Block 2

a. RR-P-1A and 1B start

b. AH-E-1A, 1B and 1C start in slow speed

c. RR-V-10A and 10B open (NOTE: These valves go shut again when RR-V-1A & 1B reach full open.)

5 seconds after block 2 initiates, block 3 initiates.

Block 3

a. DR-P-1A and 1B start

b. DR-V-1A and 1B open

c. DC-P-1A and 1B start

d. ES selected NS pumps start

e. ES selected NR pumps start and discharge valves open.

5 seconds after block 3 initiates, block 4 initiates.

Block 4

a. AH-E-15A and 15B start

b. AH-E-27A and 27B start

c. SW-P-2A and 2B start

d. SW-P-4A and 4B start

e. BS-P-1A and 1B get start permissive signal (see Figure 7)

f. EF-P-2A and 2B get start permissive signal

The following loads are also shed on a 1600# ES signal:

- SF-P-1A and 1B
- AH-E-8A and 9A
- "A" side BAMT heaters
- BAMT mixer

In addition to the above load sheds, the following equipment will trip on an ES signal.

- Non-ES selected Nuc. River pump
- SR-P-1C if SR-P-1B is running

5.3 ESAS Electrical

5.3.1 Block Loading

If offsite power is available, the timers of the automatic loading sequence will begin to time out starting ES components. The timers limit the maximum voltage dip on 4KV ES buses E and D when offsite power is available and also insure the reliability of the emergency diesel generators on loss of offsite power. Intervals for the load sequence timers are as follows: Block 1 - upon ES signal initiation with onsite power or after diesel generator 1A(B) is ready to carry load, Block 2 - five seconds after ES signal, Block 3 - ten seconds after ES signal and Block 4 - fifteen seconds after ES signal.

5.3.2 Loss of Power

Should offsite power be lost, the undervoltage relays of D and E 4160V ES buses will reset the timers of the loading sequences and clear the buses with the exception of the Block 1 load which remains in the connected position. See Figure 5. Non-essential loads will be locked-out. The 27/86 lockout relays used for this function are electrically resetable from panel PCR after the ES signal has been cleared and all circuit breaker amber disagreement lights have been reset. The 4KV feeders to the 480 volt unit substations are not tripped by 4KV bus undervoltage; however, the 480 volt loads will be tripped by the 4100 volt undervoltage relays either on low voltage or degraded voltage. As soon as the diesel generators connect to their respective buses, the block 1 safeguards load is restarted and the timers of the loading sequences block 2, 3 and 4 will time out, thus restarting these emergency loads.

The following is a listing of sequence of events for various conditions of an ES actuation and an undervoltage.

a. ES followed by undervoltage

❶ ES initiates block loading

❷ Diesel generator starts but generator breakers do not close

❸ UV trips all equipment except block 1. Non-ES equipment is locked out. Refer to Appendix A. ES prevents auto transfer of 1CES Valves MCC and DC Bus 1M. UV also starts 2.5 second timer for diesel generator breaker closure.

❹ Diesel generator breaker closing reinitiates block loading (after 2.5 second timer times out).

NOTE

If the ES system is bypassed after actuation, block loading will not occur again. Only block 1 equipment would re-start on the diesel.

The purpose of the 2.5 second timer is to allow the EMF of Block 1 motors to collapse prior to the diesel generator breaker closing.

Form ES-401-6

Written Exam Question Worksheet

Q # 037

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 001 A1.06 Page 3.1-8 Tier # 2

RO/SRO Importance Rating 4.1 4.4 Group # 1

Measurement Ability to predict/monitor changes in parameters associated with operating the CRDS controls including: Reactor power

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

The following data is obtained during an approach to criticality:

CRD Grp 5 Position	NI-11 Reading
0%	2.0 cps
25%	2.5 cps
50%	4.0 cps
75%	5.0 cps

Which ONE (1) of the following control rod positions is closest to where you would predict criticality?

A. 15% on Group 6

B. 40% on Group 6

C. 50% on Group 6

D. 95% on Group 6

Technical Reference OP 1103-8, Approach to Criticality, Rev. 46 pages 12 & 13

Open Exam Reference OP 1103-8, Approach to Criticality, Rev. 46 pages 12 & 13

Learning Objective V.I.07.04 For the 1/M plot:
c. Given plant data, construct and interpret the plot.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☐ 55.43 ☑ 55.45 .5

Discriminant Validity Statements

A Plausible, but uses second calculation.

B Plausible, but uses first calculation, starting at top of graph.

C Plausible, but connects initial and last data points.

D Correct answer.

Comments None.

AmerGen	TMI - Unit 1 Operating Procedure	Number 1103-8
Title **Approach to Criticality**		Revision No. 46
Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date 03/23/00

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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Answer Key

	Signature	Date
Originator	[illegible]	03/14/2000
Procedure Owner	[illegible]	14 March 2000
PRG	Howard Crawford PRG Vice Chairman	3/22/00
Approver	[illegible]	3/19/00

1/M = Co/C

*NI-*_____

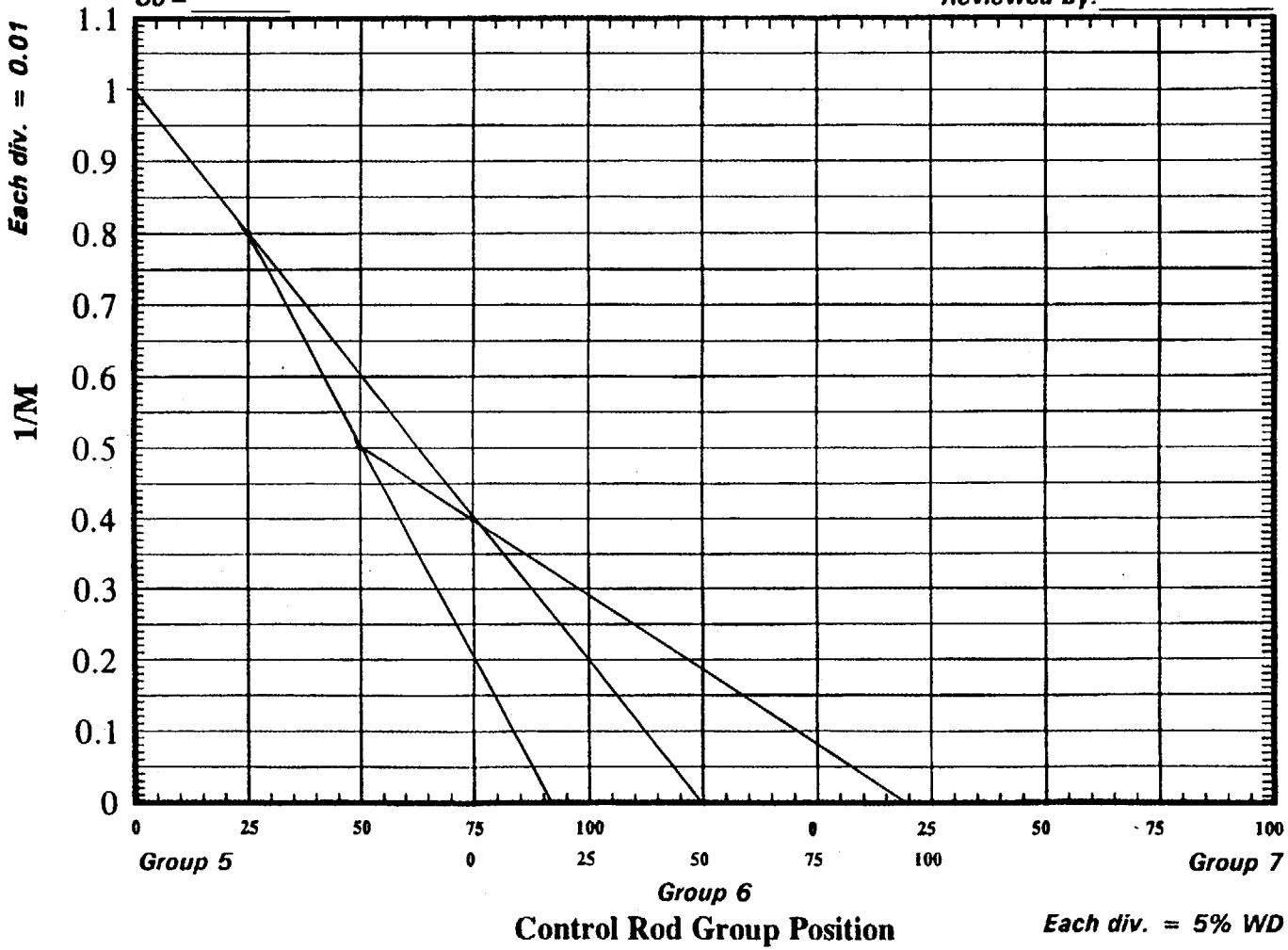
Co = _____

*Date:*_____

*Performed by:*_____

*Reviewed by:*_____

Figure 1
1/M vs CRG Position



Approach to Criticality

DATA SHEET 1

1/M Plot Data

NI- 11

C_0 = 2 CPS at 0 % withdrawn on Group 5

at _____(Date/Time). Enter C_0 data on the first line.

Rod Position (% and Group)	Stable Count Rate. C (CPS)	1/M = C_0/C	Time	Date	Initial
0% 5	2	1.0			
25% 5	2.5	0.8			
50% 5	4	0.5			
75% 5	5	0.4			

Continue on another sheet if necessary. Be sure to fill in page information.

Performed by _____ Date/Time _____

Reviewed by _____ Date _____

AmerGen	TMI - Unit 1 Operating Procedure	Number 1103-8
Title **Approach to Criticality**		Revision No. 46
Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date 03/23/00

This document is within QA plan scope	X	Yes		No
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Procedure Owner	[signature]	14 March 2000
PRG	[signature] PRG Vice Chairman	3/22/00
Approver	[signature]	3/19/00

DATA SHEET 1

1/M Plot Data

NI-_____

C_0 = _____ CPS at _____% withdrawn on Group _____

at _____(Date/Time). Enter C_0 data on the first line.

Rod Position (% and Group)	Stable Count Rate, C (CPS)	$1/M = C_0/C$	Time	Date	Initial
		1.0			

Continue on another sheet if necessary. Be sure to fill in page information.

Performed by _____ Date/Time _____

Reviewed by _____ Date _____

1/M = Co/C

*NI-*_____

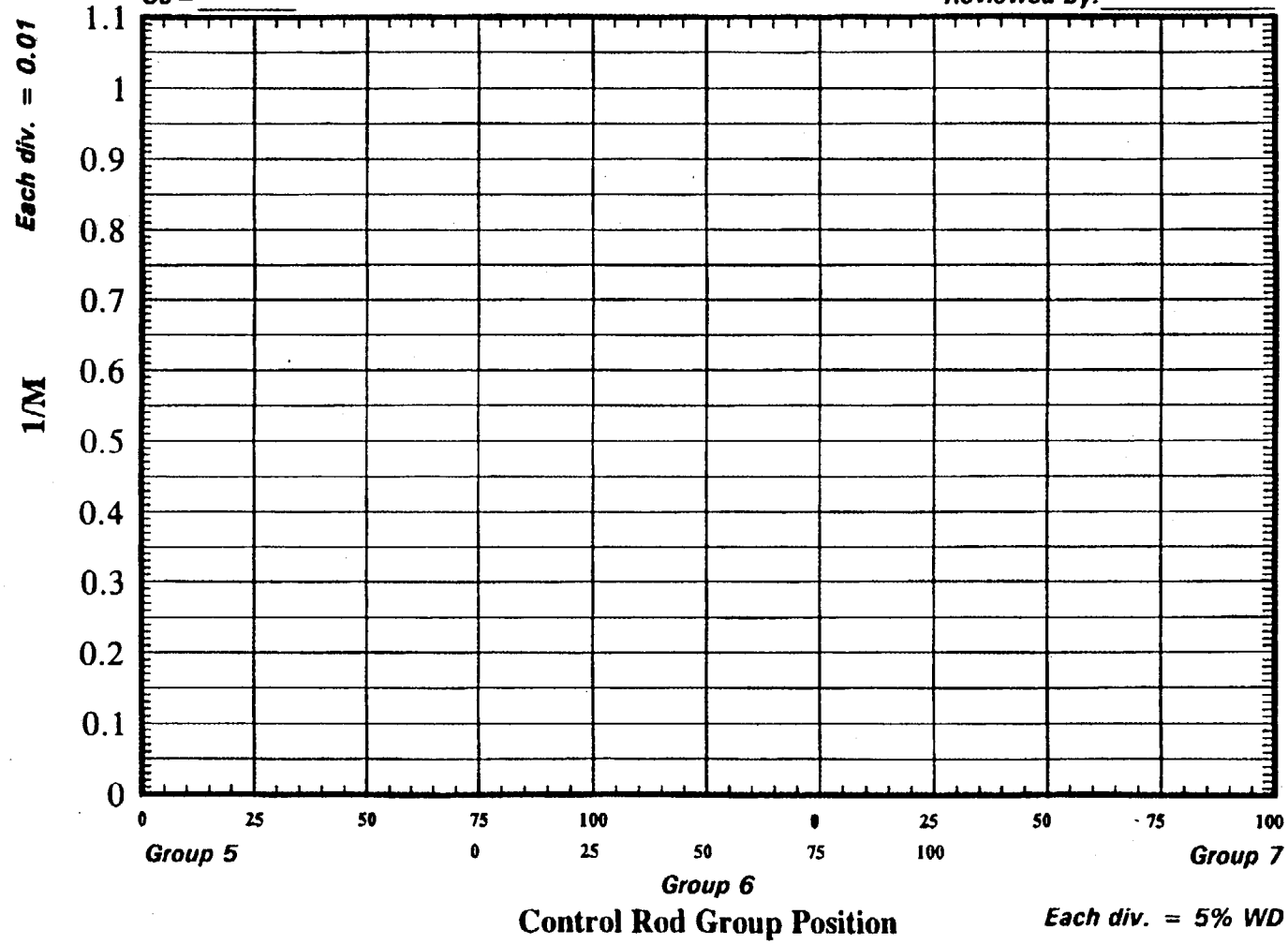
Co = _____

*Date:*_____

*Performed by:*_____

*Reviewed by:*_____

Figure 1
1/M vs CRG Position



AmerGen	TMI - Unit 1 Operating Procedure	Number **1103-8**
Title **Approach to Criticality**		Revision No. **46**
Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date 03/23/00

This document is within QA plan scope	X	Yes		No
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	Signature	Date
Originator	Gary J Dyle	03/14/2000
Procedure Owner	[illegible]	14 March 2000
PRG	Howard Crawford PRG Vice Chairman	3/22/00
Approver	[illegible]	3/19/00

DATA SHEET 1

1/M Plot Data

NI-_____

C_0 = _____ CPS at _____% withdrawn on Group _____

at _____(Date/Time). Enter C_0 data on the first line.

Rod Position (% and Group)	Stable Count Rate. C (CPS)	$1/M = C_0/C$	Time	Date	Initial
		1.0			

Continue on another sheet if necessary. Be sure to fill in page information.

Performed by _____ Date/Time _____

eviewed by _____ Date _____

1/M = Co/C

*NI-*_____

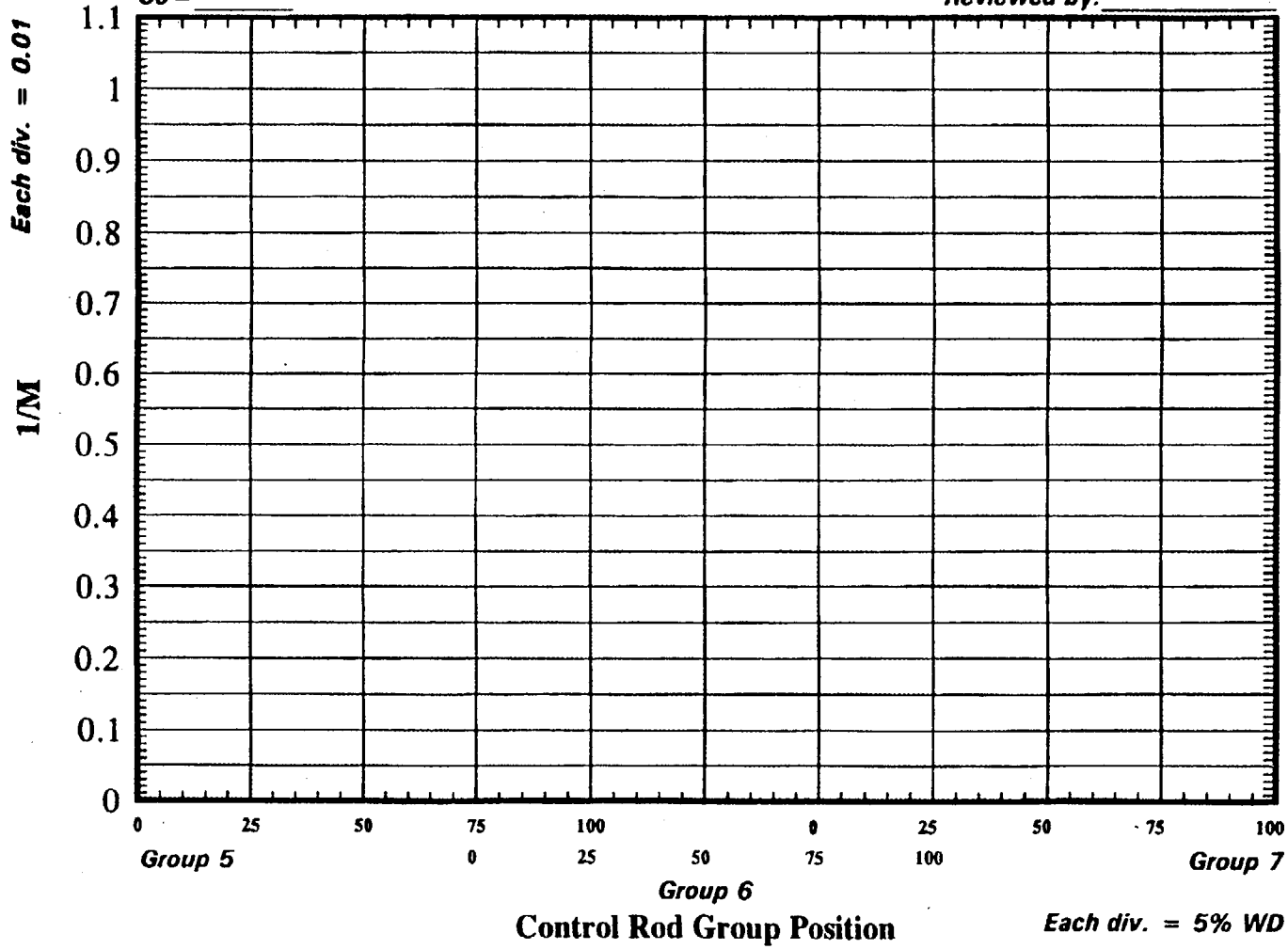
Co = _____

*Date:*_____

*Performed by:*_____

*Reviewed by:*_____

Figure 1
1/M vs CRG Position



Form ES-401-6

Written Exam Question Worksheet

Q # 038

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 001 A4.10 Page 3.1-10 Tier # 2

RO/SRO Importance Rating 3.5 3.9 Group # 1

Measurement Ability to manually operate and/or monitor in the control room: Determination of an ECP

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

Calculate an Estimated Critical Rod Position (ECP) for the following plant conditions:
- 20 hours post trip
- Plant has been running at 100% power since initial cycle startup
- 400 EFPD
- 890 ppm Boron from RCS sample
- Boron correction factor of 0.98
- CRD Group 8 at 30%
- T-ave 532°F

Which ONE (1) of the following rod tolerance bands is the most correct for this ECP?

	Minimum	Maximum
A.	72% on Group 5	55% on Group 7
B.	11% on Group 6	73% on Group 7
C.	29% on Group 6	47% on Group 7
D.	30% on Group 7	100% on Group 7

Technical Reference OP 1103-15B, Estimated Critical Conditions, Rev 29 entire procedure

Open Exam Reference 1103-15B, Estimated Critical Conditions, Rev 29 entire procedure
Calculator
Ruler

Learning Objective V.B.11.05 Prior to startup, perform independent estimated critical rod position and estimated critical boron concentration calculations in accordance with OP 1103-15B.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .5, .8

Discriminant Validity Statements

A Plausible, but uses incorrect rod worth curve.

B Correct answer.

C Plausible, but uses incorrect Xenon tolerance value.

D Plausible, but uses peak Xenon value.

Comments None.

CONTROLLED COPY

GPU NUCLEAR	TMI Operating Procedure	Number 1103-15B

Title

Estimated Critical Conditions

Revision No. 29

Applicability/Scope	Responsible Office	Effective Date
TMI Division	Plant Operations Dir.	10/16/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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Answer Key

	Signature	Date
Originator	R. Thopasso for PSB	10/16/99
Procedure Owner	R. Thopasso for PSB	10/16/99
PRG	[illegible]	10/16/99
Approver	James N. Onkes	10/16/99

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	Figure 5, Samarium and Plutonium Buildup following Reactor Shutdown	16
	Figure 6, Integral Control Rod Worth	17

1.0 **REFERENCES**

1.1 TMI-1 FSAR Section 3.2.2

1.2 TDR-1244, TMI-1 Cycle 13 Physics Data Manual

1.3 TMI-1 Technical Specifications Sections 3.5.2 and 3.1.3

1.4 Plant Process Computer (1105-10A)

1.5 Approach to Criticality (OP 1103-8)

1.6 Soluble Poison Concentration Control (OP 1103-4)

1.7 Plant Process Computer Xenon Program

1.8 Transient Monitor System (1105-15)

1.9 BWFC Operating Guidelines, 64-1234740-00, Reactivity Balance Guidelines

1.10 TMI Reactivity Management Program (1085)

0 **LIMITS AND PRECAUTIONS**

2.1 Verify that all figures used to determine core reactivities are updated to the present cycle burnup in the current fuel cycle.

2.2 Rod withdrawal sequence will be in accordance with OP 1103-8 Approach to Criticality.

> **NOTE**
>
> Shift Engineering maintains a computer program which can calculate shutdown margin and reactivity balance for INFORMATION ONLY and trending purposes. All official record calculations shall use the enclosed figures and data sheets.

2.3 All positive or negative reactivity values will be taken from the enclosed figures. Verify that the correct signs are used on each Enclosure.

> **NOTE**
>
> Report to the Shift Supervisor if, after checking the calculations, an anomaly exists. Evaluate the discrepancy and seek to resolve it.

3.0 **PROCEDURE**

3.1 Estimated Critical Rod Position

3.1.1 Purpose

The purpose of this section is to estimate the critical rod position prior to startup. It may be necessary to perform this section multiple times if plant conditions change before actual startup to account for changes in xenon or actual boron concentration.

NOTE
Obtain Enclosure 1 for computation.

3.1.2 Reference Conditions

The reference conditions for an ECP are 532°F, 0 %FP, 2155 psig, no xenon, and CRG 1-8 at 100% WD. (Ref. 1.2)

3.1.3 Calculational Method For ECP

3.1.3.1 Obtain the average reactor coolant temperature, TAVE, from the PPC or from the digital display window on the Control Room Center Console.

3.1.3.2 Obtain the cycle burnup from NAS Display 1 or the hourly log.

3.1.3.3 Obtain the latest measured boron concentration from the Reactor Coolant Chemistry Analysis, and check the Control Room log to verify that no major boron concentration changes have been made since the analysis.

3.1.3.3.1 If major boron concentration changes have been made since the latest sample, do not proceed. Request a new RCS boron concentration measurement.

3.1.3.3.2 Before continuing, ensure the RCS is in boron concentration equilibrium. If necessary, request additional samples until two consecutive measurements are within 10 ppm of each other and no additional changes have been made to RCS boron.

3.1.3.3.3 Adjust the measured boron concentration to account for B-10 depletion using the boron correction factor (NAS Display 10, Control Room Log, or Nuclear Engineering).

3.1.3.4 Record the expected CRG 8 position at criticality.

3.1.3.5 Determine the fuel excess reactivity per Figure 1.

3.1.3.6 Determine the CRG 8 reactivity worth of Section 3.1.3.4 per Figure 2.

3.1.3.7 Determine the Hot Zero Power (HZP) Inverse Boron Worth associated with the cycle burnup per Figure 3.

3.1.3.8 Determine the reactivity associated with the boron concentration by dividing the current concentration (Step 3.1.3.3) by the HZP Inverse Boron Worth (Step 3.1.3.7).

3.1.3.9 Obtain the xenon worth from NAS Display 22.

3.1.3.9.1 If this program is not available, obtain a value from the program XENC# (where "#" is the cycle number). Figure 4 may be used if Plant Process Computer Program and Nuclear Engineering are unavailable provided that power was constant (within ± 2% FP) for at least the last 40 hours prior to shutdown.

3.1.3.10 Determine the reactivity associated with samarium and plutonium buildup after shutdown by using Figure 5.

3.1.3.11 Determine the inserted CRG 5-7 worth required to be critical.

> **NOTE**
>
> Inserted rod worth is used because the reference condition is for an ARO configuration.

3.1.3.12 Determine the estimated critical rod position per Section 3.1.3.11 and Figure 6.

3.1.3.13 Determine the critical rod position tolerance band for either a xenon transient or steady state xenon situation.

> **NOTE**
>
> For the purposes of this procedure, xenon is defined as being in steady state when the xenon worth is between zero and -0.5% $\Delta k/k$; transient xenon is when xenon worth is more negative than -0.5% $\Delta k/k$.

3.1.3.14 Sign the appropriate location on Enclosure 1 and forward it to the appropriate operations person for filing.

3.1.3.15 Send a copy of the completed Enclosure 1 to Manager, Shift Engineering.

3.2 Estimated Critical Boron Concentration

3.2.1 Purpose

The purpose of this section is to estimate the critical boron concentration prior to startup. This portion of the procedure need only be performed to obtain a critical boron concentration. The RCS is then deborated (prior to withdrawing CRG 5-7) to this concentration, and CRG 5-7 are withdrawn in accordance with OP 1103-8 Approach to Criticality.

NOTE

Obtain Enclosure 2 for ECB Calculation.

3.2.2 Reference Conditions

The reference conditions for an ECB are 532°F, 0 %FP, 2155 psig, no xenon, and CRG 1-8 at 100% WD.

3.2.3 Calculational Method for ECB

3.2.3.1 Obtain the average reactor coolant temperature, TAVE, from the PPC or from the digital display window on the Control Room Center Console.

3.2.3.2 Obtain the cycle burnup from NAS Display 1 or the hourly log.

3.2.3.3 Obtain the latest measured boron concentration from the Reactor Coolant Chemistry Analysis. and check the Control Room log to verify that no major boron concentration changes have been made since the analysis.

NOTE

No B-10 correction factor is used for the measured boron in #2 of Enclosure 2 since actual concentration is required to determine water addition amounts. B-10 depletion is accounted for in #11 of Enclosure 2.

3.2.3.3.1 If major boron concentration changes have been made since the latest sample, request a new RCS boron concentration measurement.

3.2.3.3.2 Until the new boron concentration is available, use 1103-4, Soluble Poison Concentration Control to estimate the current boron concentration to calculate a preliminary ECB.

3.2.3.4 Record the desired critical rod positions.

NOTE

Normally for a xenon free core, the desired critical rod position will be between 75% WD on CRG-6 and 25% WD on CRG-7. This will allow a power increase to 90% F.P. using CRG-6 and CRG-7. With CRG-7 at approximately 90% WD, it will be necessary to feed and bleed for control of equilibrium xenon buildup.

3.2.3.5 Determine the fuel excess reactivity per Figure 1.

3.2.3.6 Determine the CRG 5-7 reactivity worth of Section 3.2.3.4 per Figure 6.

3.2.3.7 Determine the CRG 8 reactivity worth of Section 3.2.3.4 per Figure 2.

3.2.3.8 Obtain the xenon worth from NAS Display 22.

3.2.3.8.1 If this program is not available, obtain a value from the program XENC# (where "#" is the current cycle number). Figure 4 may be used if Plant Process Computer Program and Nuclear Engineering are unavailable provided that power was constant (within ± 2% FP) for at least the last 40 hours prior to shutdown.

3.2.3.9 Determine the reactivity associated with samarium and plutonium buildup after shutdown by using Figure 5.

3.2.3.10 Determine the boron reactivity worth required for criticality.

3.2.3.11 Determine the HZP Inverse Boron Worth associated with the cycle burnup per Figure 3.

3.2.3.12 Determine the corrected Critical Boron Concentration by multiplying the required boron worth (Step 3.2.3.10) by the HZP Inverse Boron Worth (Step 3.2.3.11).

NOTE
11a is the Critical Boron Concentration based on undepleted boron.

3.2.3.13 Determine the Final Mixed Boron depletion correction factor based on current correction factor and accounting for predicted boron additions.

NOTE
Only boron additions will affect the depletion correction factor.

3.2.3.14 Determine the Estimated Measured Critical Boron Concentration by dividing the corrected CBC (Step 3.2.3.12) by the depletion correction factor (Step 3.2.3.13).

3.2.3.15 Sign the appropriate location on Enclosure 2 and forward it to the appropriate operations person for filing.

3.2.3.16 Send a copy of the completed Enclosure 2 to Manager, Shift Engineering.

3.3 Actual Subcriticality Calculation (ASC)

NOTE
Obtain Enclosure 3 for ASC calculation.

NOTE
This calculation is only valid at 532° F and 0% FP conditions.

3.3.1 Record the present control rod positions.

3.3.2 Determine the Integral Control Rod Worth for the present control rod position using Figure 6.

3.3.3 Perform an Estimated Critical Rod Position per Section 3.1.

3.3.4 Record the "Inserted CRG 5-7 Worth Required for Criticality" (#10 of Enclosure 1).

3.3.5 Calculate the Actual Subcriticality.

3.3.6 Sign the appropriate location on Enclosure 3 and forward to the appropriate Operations person for filing.

3.3.7 Send a copy of completed Enclosure 3 to Manager, Shift Engineering.

ENCLOSURE 1

Estimated Critical Rod Position

I. CALCULATION IS FOR AN ECP AT 532 ± 2°F ON: DATE _____ TIME _____

1. CYCLE BURNUP 400 EFPD
2.a. FINAL MEASURED BORON CONCENTRATION 890 ppmB
2.b. BORON DEPLETION CORRECTION FACTOR 0.98
(NAS Display 10, Control Room Log, Nuclear Engineering)
2.c. FINAL CORRECTED BORON CONCENTRATION (2a)X(2b) = 872.2 ppmB
3. CRG 8 POSITION AT CRITICALITY 30 % WD
4. FUEL EXCESS REACTIVITY (FIG 1) 10.44 % Δk/k
5. CRG 8 REACTIVITY WORTH (FIG 2) -0.12 % Δk/k
6. INVERSE BORON WORTH (FIG 3) 135.8 ppmB/% Δk/k
7. BORON REACTIVITY WORTH
[ppmB #2c / Inverse Boron #6] x (−1) = -6.42 % Δk/k
8. XENON REACTIVITY WORTH (PPC. NUCLEAR ENGR., FIG 4) -2.66 % Δk/k
9. SAMARIUM AND PLUTONIUM BUILDUP (FIG 5)
9a. TIME SINCE LAST SHUTDOWN 20 HRS
9b. REACTIVITY DUE TO BUILDUP -0.01 % Δk/k
10. INSERTED CRG 5-7 WORTH REQUIRED FOR CRITICALITY
[(FUEL) #4 + (CRG 8) #5 + (BORON) #7 + (XENON) #8 + (SM) #9b] x (−1) = -1.23 % Δk/k
11. ESTIMATED CRITICAL ROD POSITION (FIG 6) 1 % WD on CRG 7
12. CRITICAL ROD POSITION TOLERANCE BAND (FIG 6)
12a. CHECK ONE
_____ 0.5% Δk/k If |Xenon (#8)| is ≤ 0.5%Δk/k)
✓ 0.8% Δk/k If |Xenon (#8)| is > 0.5%Δk/k)
12b. MINIMUM ROD WITHDRAWAL LIMIT (#10 - #12a) 9 % WD on CRG 6
12c. MAXIMUM ROD WITHDRAWAL LIMIT (#10 + #12a) 71 % WD on CRG 7

CALCULATED BY: _____ DATE/TIME _____

APPROVED BY (SRO): _____ DATE/TIME _____

Send copy of this Enclosure to Manager, Shift Engineering
Send original to Operations for filing

ENCLOSURE 2

Estimated Critical Boron Concentration

I. CALCULATION IS FOR AN ECP AT 532 ± 2°F ON: DATE _____ TIME _____

1. CYCLE BURNUP _____EFPD
2. PRESENT MEASURED BORON CONCENTRATION _____ppmB
3. DESIRED CRITICAL ROD POSITION
   - 3a. CRG 1-4 __100__% WD
   - 3b. CRG 5 _____% WD
   - 3c. CRG 6 _____% WD
   - 3d. CRG 7 _____% WD
   - 3e. CRG 8 _____% WD
4. FUEL EXCESS REACTIVITY (FIG 1) _____% Δk/k
5. DESIRED CRITICAL CRG 5-7 REACTIVITY WORTH (FIG 6) _____% Δk/k
6. DESIRED CRITICAL CRG 8 REACTIVITY WORTH (FIG 2) _____% Δk/k
7. XENON REACTIVITY WORTH (PPC, NUCLEAR ENGR., FIG 4) _____% Δk/k
8. SAMARIUM AND PLUTONIUM BUILDUP (FIG 5)
   - 8a. TIME SINCE SHUTDOWN _____HRS
   - 8b. REACTIVITY DUE TO BUILDUP _____% Δk/k
9. BOREON REACTIVITY WORTH REQUIRED FOR CRITICALITY

$$\left[\underset{\#4}{(\text{FUEL})} + \underset{\#5}{(\text{CRG } 5-7)} + \underset{\#6}{(\text{CRG } 8)} + \underset{\#7}{(\text{Xenon})} + \underset{\#8b}{(\text{SM})}\right] \times (-1) =$$ _____% Δk/k

10. INVERSE BORON WORTH (FIG 3) _____ppmB/% Δk/k
11. CRITICAL BORON CONCENTRATION
    - 11.a CORRECTED CRITICAL BORON CONCENTRATION

$$\left[\underset{\#10}{(\text{Inverse Boron})} \times (-1) \times \underset{\#9}{(\text{Boron React})}\right] =$$ _____ppmB

    - 11.b FINAL MIXED BORON DEPLETION CORRECTION FACTOR (NAS Display 10, Control Room Log, Nuclear Engineering) _____
    - 11.c ESTIMATED MEASURED CRITICAL BORON CONCENTRATION (11a) / (11b) =_____ppmB

CALCULATED BY: _____ DATE/TIME _____

APPROVED BY (SRO): _____ DATE/TIME _____

nd copy of this Enclosure to Manager, Shift Engineering
nd original to Operations for filing

ENCLOSURE 3

Actual Subcriticality Calculation

I. CALCULATION IS FOR ASC AT: DATE_____TIME_____

1. PRESENT ROD POSITIONS

1a.	CRG 1-4	100	% WD
1b.	CRG 5	_____	% WD
1c.	CRG 6	_____	% WD
1d.	CRG 7	_____	% WD

2. INTEGRAL CONTROL ROD WORTH (Fig. 6) _____% Δk/k

3. INSERTED CRG 5-7 WORTH REQUIRED FOR CRITICALITY (ENCLOSURE 1, #10) _____% Δk/k

4. ACTUAL SUBCRITICALITY (#2 - #3) _____% Δk/k

CALCULATED BY: _____ DATE/TIME _____

APPROVED BY (SRO): _____ DATE/TIME _____

Send copy of this Enclosure to Manager, Shift Engineering
Send original to Operations for filing

FIGURE 1
Cycle 13 Fuel Excess Reactivity

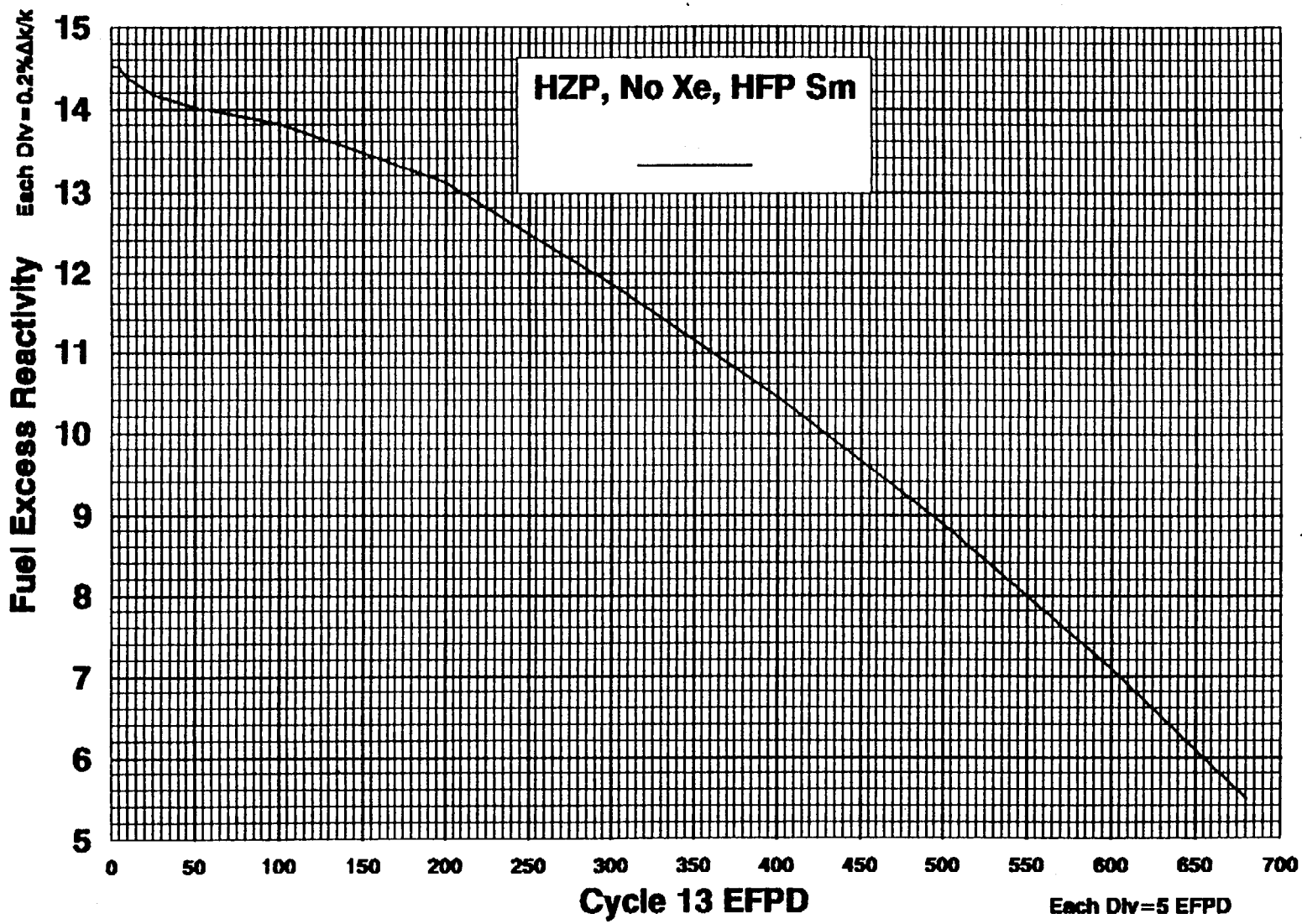
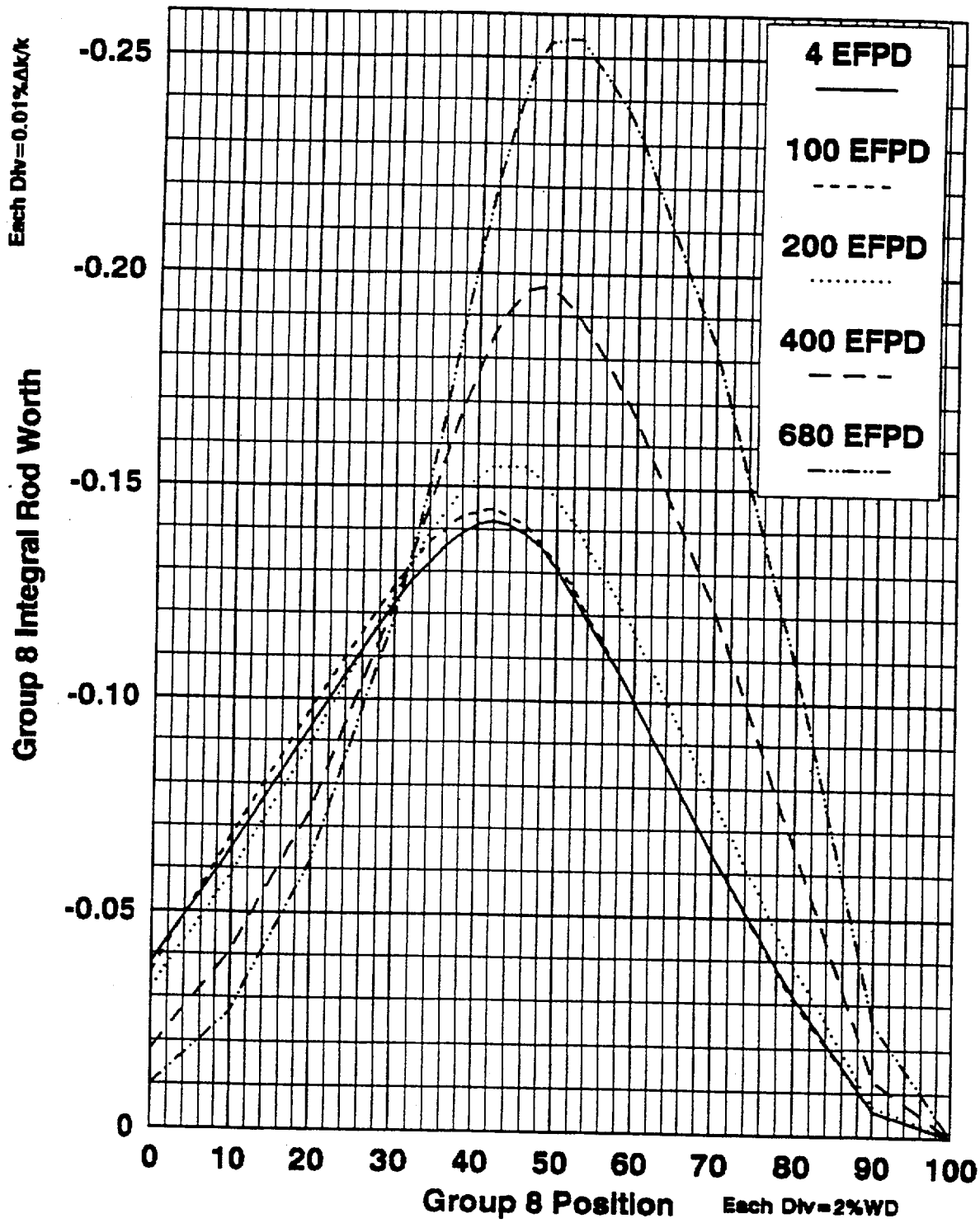


FIGURE 2

Group 8 Integral Rod Worth
Cycle 13, HZP, No Xenon



NOTE
Linearly interpolate between EFPD Curves.

FIGURE 3
Cycle 13 HZP Inverse Boron Worth

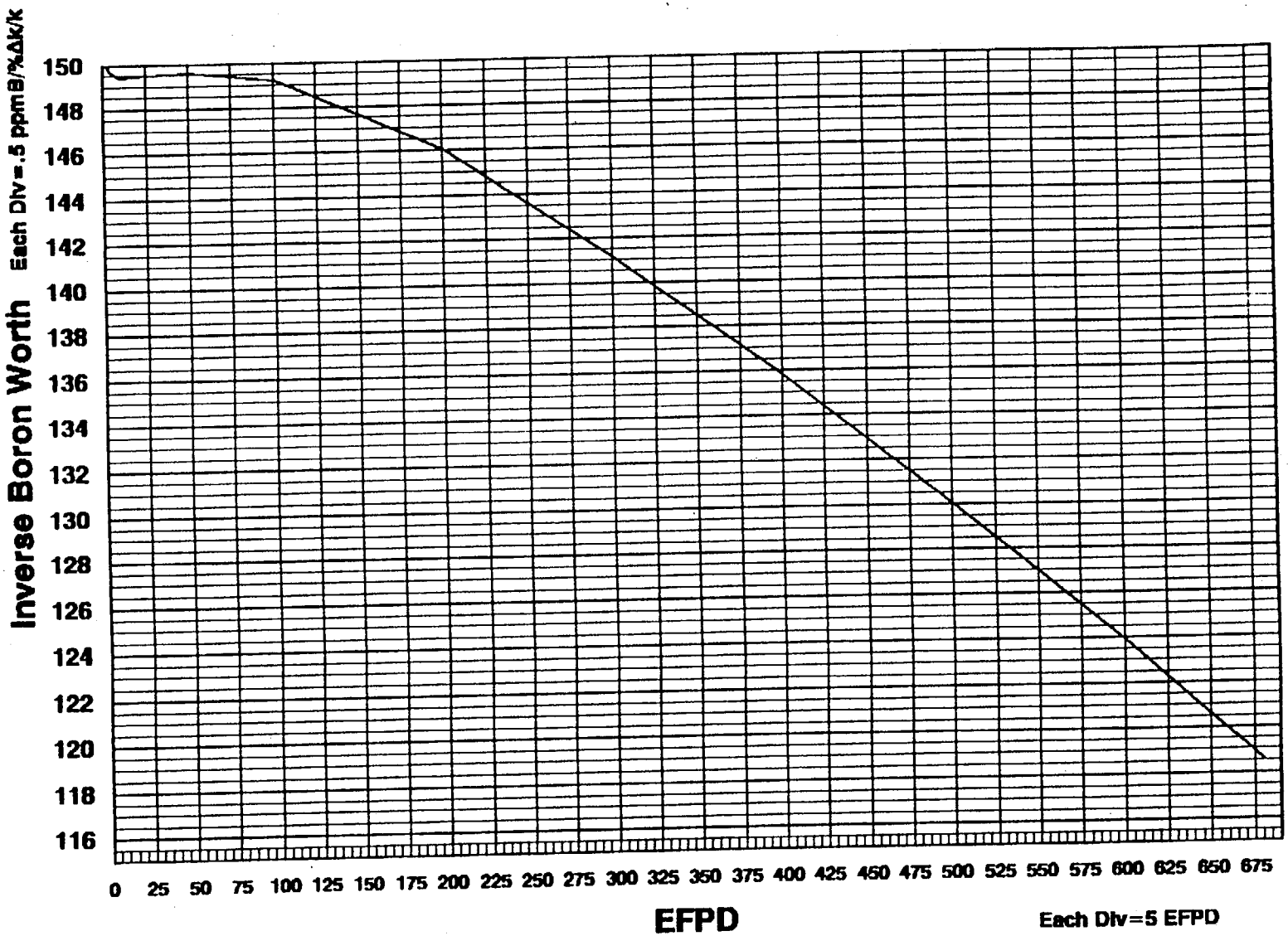


FIGURE 4
Cycle 13 Transient Xenon Reactivity Worth

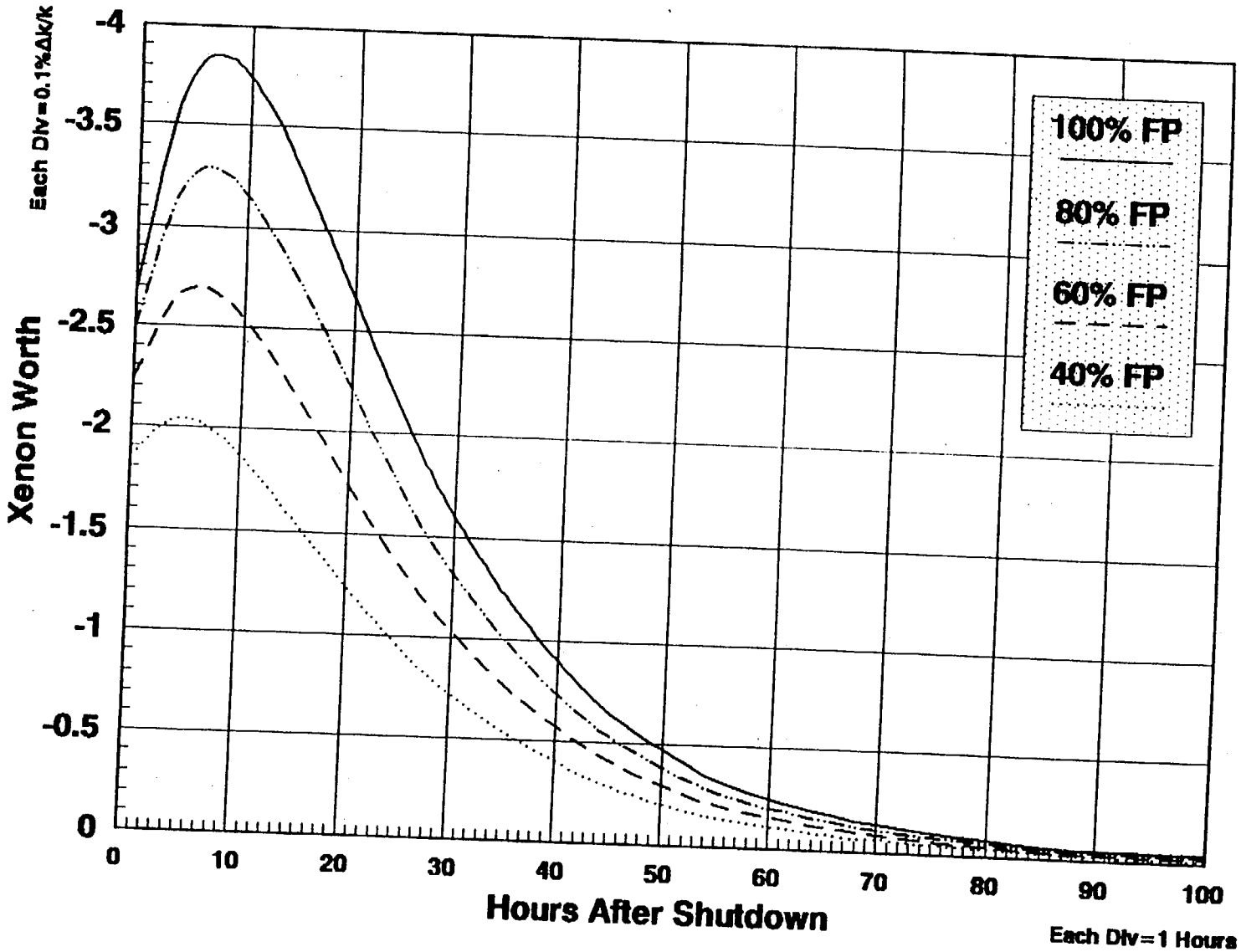
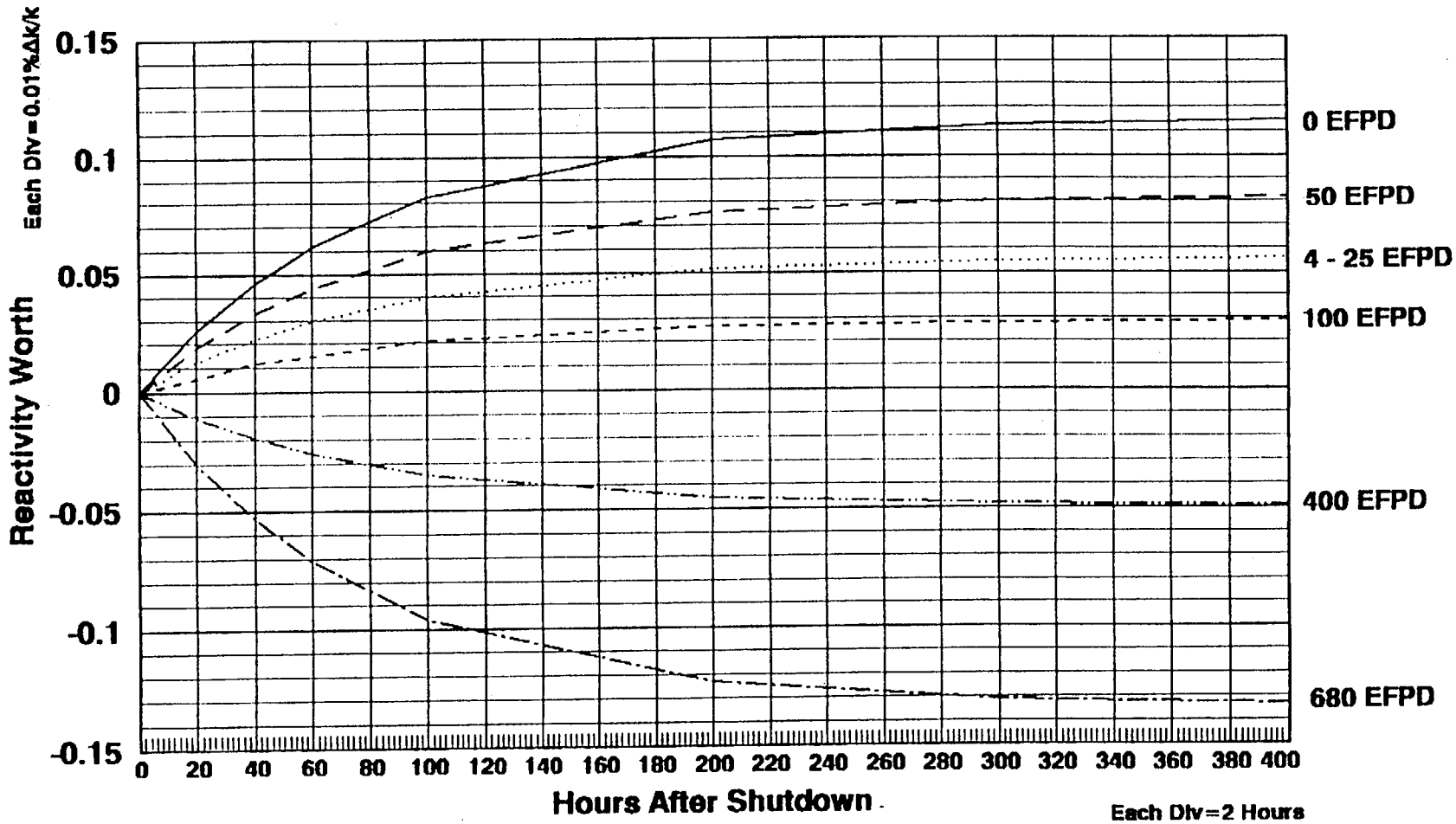
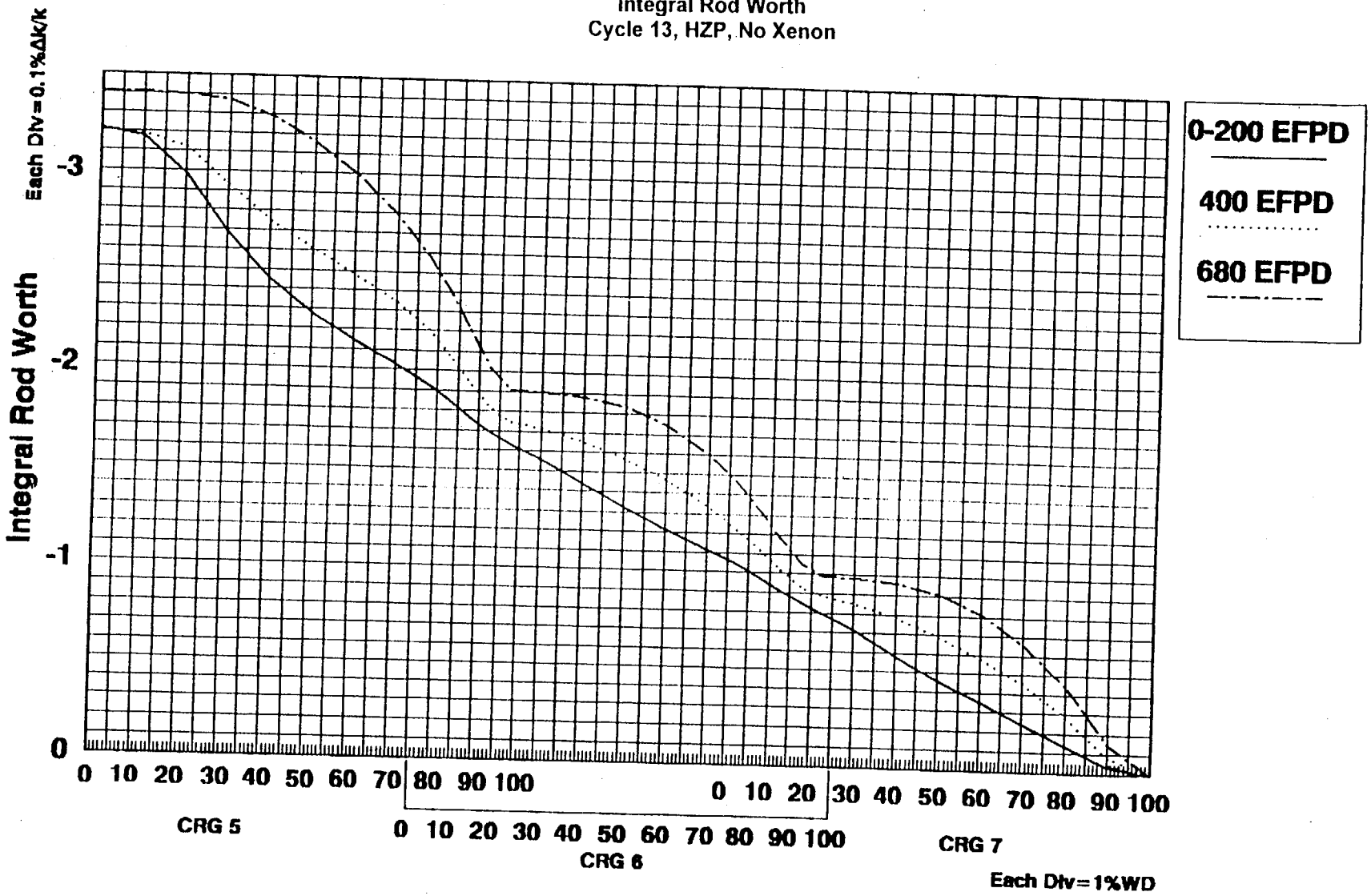


FIGURE 5
Cycle 13 Samarium and Plutonium Buildup
Following Reactor Shutdown



*NOTE: Linearly interpolates between EFPD.
If startup at anytime during cycle is within 5 days of a previous startup, contact Nuclear Engineering for the appropriate reactivity worth.

FIGURE 6
Integral Rod Worth
Cycle 13, HZP, No Xenon



NOTE

Linearly interpolate between EFPD Curves.

Form ES-401-6

Written Exam Question Worksheet

Q # 039

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 003 K5.04 Page 3.4-7 Tier # 2

RO/SRO Importance Rating 3.2 3.5 Group # 1

Measurement Knowledge of the operational implications of the following concepts as they apply to the RCPs: Effects of RCP shutdown on secondary parameters, such as steam pressure, steam flow, and feed flow

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

The following plant conditions exist:
- Reactor power is 70%
- 4 RCPs in operation
- ICS is in full AUTO
- High vibrations exist on RC-P-1B

Which ONE of the following describes the correct response of the feedwater system when RC-P-1B is secured?

A. Feed flow to OTSG A increases
Feed flow to OTSG B increases

B. Feed flow to OTSG A increases
Feed flow to OTSG B decreases

C. Feed flow to OTSG A decreases
Feed flow to OTSG B decreases

D. Feed flow to OTSG A decreases
Feed flow to OTSG B increases

Technical Reference DWG D55731 Rev. Q

Open Exam Reference

Learning Objective IV.E.27.25 Given the following plant conditions:
a. Varying RC flow (or)
b. Plugging and/or fouled OTSG tubes,
Predict and explain the effects that the delta-Tc circuit would have on total feedwater demand, A OTSG feedwater demand and B OTSG feedwater demand.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☐ 55.43 ☑ 55.45 .7

Discriminant Validity Statements

A Plausible if Delta T-cold ratio operation is misunderstood.

B Plausible if Delta T-cold ratio operation is misunderstood.

C Plausible if Delta T-cold ratio operation is misunderstood.

D Correct answer.

Comments None.

Form ES-401-6

Written Exam Question Worksheet

Q # 040

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 003 A1.02 Page 3.4-8 Tier # 2

RO/SRO Importance Rating 2.9 2.9 Group # 1

Measurement Ability to predict/monitor changes in parameters associated with operating the RCPs controls including: RCP pump and motor bearing temperatures

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

Which ONE (1) of the following sets of Reactor Coolant Pump motor temperatures would require a power reduction and trip of the affected pump?

A. Motor stator - 110°C
Motor radial bearings - 170°F
Motor thrust bearing - 185°F

B. Motor stator - 120°C
Motor radial bearings - 190°F
Motor thrust bearing - 180°F

C. Motor stator - 130°C
Motor radial bearings - 150°F
Motor thrust bearing - 175°F

D. Motor stator - 140°C
Motor radial bearings - 170°F
Motor thrust bearing - 170°F

Technical Reference AbP 1203-16, Reactor Coolant Pump and Motor Malfunction, Rev. 40 page 4 & 5

Open Exam Reference

Learning Objective V.C.06.03 State the immediate manual actions required for the following RCP malfunctions per AP 1203-16:
b. Loss of RCP motor cooling water (NSCCW)

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☐ 55.43 ☑ 55.45 .5

Discriminant Validity Statements

A Plausible due to high thrust bearing temperature that may be mistaken for radial bearing temperature requirement.

B Correct answer.

C Plausible due to high radial bearing temperature

D Plausible due to high stator temperature

Comments None.

FOR INFORMATION ONLY

GPU NUCLEAR	TMI Abnormal Procedure	Number 1203-16
Title: **Reactor Coolant Pump and Motor Malfunction**		Revision No. 40
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: 10/12/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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8	40						

	Signature	Date
Originator	R S Campbell FOR F. KALINKO	10-12-99
Procedure Owner	R S Campbell	10-12-99
PRG	[illegible]	10/12/99
Approver	James Powell	10/12/99

2. Increased monitoring of SLO flow is performed as specified by the Plant Operations Director.

3. Additional precautions are taken to insure that a standby makeup pump is available to provide seal injection in the case where the operating makeup pump would trip.

c. RC Drain Tank level control considerations may restrict continued operation with No. 2 Seal degradation. Limiting aspects of RC Drain Tank control include level, temperature and maintaining additional tank capacity to receive what must be rejected from the RC Drain Tank.

d. If No. 1 Seal leakoff is low and below the recommendations of OP 1103-6, it may be indicative of either an abnormally tight No. 1 seal or most likely an abnormally open No. 2 Seal. A bucket check of No. 1 seal leakoff per OP 1103-6 is recommended under these circumstances. Continued operation of the RCP within the guidance provided above can be expected.

2.0 LOSS OF RCP MOTOR COOLING WATER

2.1 Symptoms

1. Loss of N.S.

a. Motor air temp. Hi computer alarm Pts. A0699, A0705, A0711, A0717.

b. RCP stator temp. Hi computer alarm Pts. A0704, A0710, A0716, A0722.

c. Motor upper guide bearing temp. Hi computer alarm Pts. A0702, A0708, A0714, A0720. Motor lower guide bearing temp. Hi computer alarm Pts. A0703, A0709, A0715, A0721.

d. Motor up thrust bearing temp. Hi computer alarm Pts. A0701, A0707, A0713, A0719. Motor down thrust bearing temp. Hi computer alarm Pts. A0700, A0706, A0712, A0718.

2.2 Immediate Action

A. Automatic Action

None

B. Manual Action

1. On loss of N.S., if any of the following temperatures are exceeded, reduce power to 50-75 Percent and trip the affected pump.

a. Motor Stator > 150°C. (302°F).

b. Motor Radial Bearings > 185°F.

c. Motor Thrust Bearings > 195°F.

2.3 Follow-up Action

Objective: To prevent damage to RCP due to excessive temperature.

1. Re-establish cooling water flow.

3.0 PUMP AND MOTOR VIBRATION

3.1 Symptoms

> **NOTE**
>
> Pump shaft vibration alarms are Alert: 15 mils, Danger: 20 mils. Motor stand vibration alarms are Alert: 3 mils, Danger: 7 mils. Setting the toggle switch to "Trip Multiply" on PLF doubles the alarm limits and causes the red bypass lights to flash.

a. Excessive pump shaft or motor stand vibration indicated on PLF.

b. High vibration computer alarm points L3122, L3123, L3124, L3125.

3.2 Immediate Action

A. Automatic Action

1. None

B. Manual Action

1. Reset alarms:

a. Reset for motor stand vibration on Bently Nevada System on PLF.

b. Reset for shaft vibration on Bently Nevada System on PLF.

2. If vibration alarms reoccur immediately, reduce power to 50 - 75 Percent and trip affected RCP.

3.3 Follow-up Action

Objective: To prevent damage to RCP due to excessive vibration.

1. Determine cause and repair.

Form ES-401-6

Written Exam Question Worksheet

Q # 041

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 004 K2.06 Page 3.1-13 Tier # 2

RO/SRO Importance Rating 2.6 2.7 Group # 1

Measurement Knowledge of bus power supplies to the following: Control instrumentation

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

Initial Conditions:
- Plant is at 100% power
- All ICS H/A stations are in AUTO

Which ONE (1) of the following condition(s) would exist on the loss of bus ATA?

A. MU-V-5 controller fails to the mid position

B. FW-V-17A/B controller fails to mid position.

C. Pressurizer heater control fails to 50% demand.

D. MS-V-4A/B control transfers to the Back-up Loaders.

Technical Reference EP 1202-42, Total or Partial Loss of ICS/NNI Auto Power, Rev. 34 page 2.

Open Exam Reference

Learning Objective V.D.22.04 Given a list of symptoms, identify the event, determine what immediate automatic actions will occur and what immediate manual actions are required IAW EP 1202-42.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☐ 55.45

Discriminant Validity Statements

A Plausible if misconception exists as to power supply for the controller.

B Plausible if misconception exists as to power supply for the controller.

C Plausible if misconception exists as to power supply for the controller.

D Correct answer.

Comments None.

FOR INFORMATION ONLY

GPU NUCLEAR	TMI Emergency Procedure	Number 1202-42
Title: **Total or Partial Loss of ICS/NNI Auto Power**		Revision No. 34
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: 10/12/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

Q-41

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12	34						
13	34						

	Signature	Date
Originator	George J. Bode (For F.J. Kacinko)	10/12/1999
Procedure Owner	George J. Bode	10/12/1999
PRG	[illegible]	10/12/99
Approver	James Powell	10/12/99

NOTE

If a partial loss of auto power occurs, not all of the symptoms and automatic actions will occur.

1.0 SYMPTOMS

1. "ICS/NNI POWER LOST" alarm, H-1-8.
2. "ICS AUTO" light and/or "SUBFEEDS - AUTO/HAND" subfeed light on ICS/NNI power monitor (Panel PCL) are off. If only a partial loss of auto power occurs, only one of the lights will go out.
3. Indications listed in Table 1 will fail. For a partial loss of auto power only the indications powered by the applicable subfeed will fail.
4. All ICS/NNI H/A station indicators (Measured Variable position) will fail. POS position remains operable (except for Turbine Control H/A Station).
5. All ICS/NNI H/A station lamps will go off.
6. All the backlighted instrument selector pushbuttons will go off.
7. Invalid and inoperable alarms are listed in Table 2A and 2B.

2.0 IMMEDIATE ACTION

2.1 Automatic Action

1. Control transferred to Hand on all H/A stations. Final control element H/A stations remain operable and retain the last demand value.

NOTE

FW control (pumps and valves) is transferred to "manual". If the reactor trips, feedwater must be reduced manually.

2. Signals transfer to provide valid main feedwater flow indication (recorder) and valid feedwater valve ΔP indication.
3. MU-V-1A/B and MU-V-3 go closed due to auto powered temperature interlocks.
4. All pressurizer heater controls are "Off " due to pressurizer lo-lo level interlock.
5. MU-V-8 travels to "thru" position due to MU tank level interlock.
6. PORV (RC-RV-2) will fail closed but is operable with manual control switch.
7. Pressurizer Spray Valve (RC-V-1) will fail closed in Auto, but is operable in Manual mode.
8. MS-V-4A/B will transfer to back-up manual loader ("BACKUP CTRL" Bailey Stations)

Form ES-401-6

Written Exam Question Worksheet

Q # 042

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 004 K5.07 Page 3.1-14 Tier # 2

RO/SRO Importance Rating 2.8 3.2 Group # 1

Measurement Knowledge of the operational implications of the following concepts as they apply to the CVCS: Relationship between SUR and reactivity

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

The following conditions exist:
- Reactor is at 10e-8 amps and stable
- RCS boron concentration is at 950 ppm
- 'A' RCBT boron concentration is at 12 ppm
- 'B' RCBT boron concentration is at 945 ppm
- 'C' RCBT boron concentration is at 1400 ppm

Makeup is initiated to raise level in MU-T-1. Two minutes later, the following indications are observed:
- NI-3 and NI-4 indicate a 0.1 DPM startup rate
- NI-3 and NI-4 level indication is rising

Which ONE (1) of the following would result in the above plant response?

A. Makeup from the BWST.

B. Makeup from the 'A' RCBT

C. Makeup from the 'B' RCBT

D. Makeup from the 'C' RCBT

Technical Reference AbP 1203-10, Unanticipated Criticality, page 2 rev. 18.

Open Exam Reference

Learning Objective V.I.07.11 Describe the causes of uncontrolled reactivity additions which have occurred in the industry or may occur at TMI including:
f. Inadvertent moderator dilution.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☐ 55.43 ☑ 55.45 .7

Discriminant Validity Statements

A Plausible since this is a potential source of makeup to the RCS.

B Correct answer.

C Plausible since this will result in a reduction of boron concentration, but not enough in the 2 minute time frame.

D Plausible since this is a potential source of makeup to the RCS.

Comments None.

FOR INFORMATION ONLY

GPU NUCLEAR	TMI Abnormal Procedure	Number Q-42 1203-10

Title

Unanticipated Criticality

Revision No. **18**

Applicability/Scope	Responsible Office	Effective Date
TMI Division	Plant Operations Dir.	12/15/98

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	[signature]	10/28/98
Procedure Owner	L. G. Nash for T. H. Good Lawrence	12-10-98
PRG	[signature]	12-10-98
Approver	L. G. Nash	12-10-98

1.0 **SYMPTOMS**

1. Unanticipated increase in neutron count rate.

2. Possible actuation of CRD outmotion inhibit alarm G-2-3 (intermediate range set at 3 DPM, resets at 0.5 DPM; source range set at 2 DPM, resets at 0.5 DPM).

3. Shutdown margin or amount plant is subcritical is determined to be less than 1% Δk/k, except during planned evolutions.

2.0 **IMMEDIATE ACTION**

2.1 Automatic Action

1. Withdrawal of control rods stop (only if alarm G-2-3, CRD outmotion inhibit actuated)

2.2 Manual Action

1. Immediately start insertion of control rods (if withdrawn) or emergency borate.

2. Continuously monitor NI 11 and 12 count rate, and NI 3 and 4 amps until the core has been verified to be shutdown by at least 1 percent Δk/k in accordance with 1103-15A, Shutdown Margin and Reactivity Balance.

3. IF high startup rate is exhibited on source and/or intermediate range indicators. THEN trip the reactor and refer to 1210-1.

* 4. Ensure there is no dilution flow into the Reactor Coolant System, or stop dilution if in progress.

* 5. Check RCS Temp - Tc and Th and stabilize temperature if possible.

3.0 **FOLLOW-UP ACTION**

OBJECTIVE

To reestablish core condition of 1 percent Δk/k shutdown.

_____ 1. Reverify steps in manual action marked with an *. Use redundant indication if possible.

_____ 2. Verify shutdown by at least 1 percent Δk/k per OP 1103-15A. Start boration if necessary.

_____ 3. Obtain RCS, makeup tank, and pressurizer sample for boron analysis.

_____ 4. Determine and correct cause for unanticipated criticality. Maintain core subcritical by at least 1% Δk/k until all evaluations are complete.

_____ 5. Refer to the Reportability Procedure AP 1044 for guidance on reportability to the NRC.

Form ES-401-6

Written Exam Question Worksheet

Q # 043

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 013 K3.03 Page 3.2-25 Tier # 2

RO/SRO Importance Rating 4.3 4.7 Group # 1

Measurement Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: Containment

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

The following plant conditions exist:
- The plant is at 100% power.
- An inadvertent "B" train 30# ESAS actuation has occurred
- The "B" train 30# ESAS actuation cannot be reset

Which ONE of the following are the direct consequences of this ESAS actuation?

A. High Reactor Coolant Pump Seal temperatures
High Control Rod Drive Stator temperatures

B. High Reactor Coolant Pump Seal temperatures
Loss of Reactor Coolant Pump Seal Injection

C. High Reactor Coolant Pump Motor temperatures
High Control Rod Drive Stator temperatures

D. High Reactor Coolant Pump Motor temperatures
Loss of Reactor Coolant Pump Seal Injection

Technical Reference OP 1105-3, Safeguards Actuation System, Rev. 41 page 22
EP 1202-8, CRD Equipment Failure, Rev. 53 page 26
AbP 1203-16, Reactor Coolant Pump and Motor Malfunction, Rev. 40 page 4

Open Exam Reference

Learning Objective IV.B.05.10/ IV.B.03.06 Identify/describe the effects of an ESAS actuation (1600#, 4#, 30#) on the NSCCW system, with or without an undervoltage condition on the ES buses.
Describe the effects of an ESAS actuation (both with and without undervoltage conditions) on the Intermediate Closed Cooling Water system.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .6

Discriminant Validity Statements

A Plausible because CRDs and RCP thermal barrier heat exchangers lose cooling water due to containment isolation of ICCW, however RCP seal injection will prevent high RCP seal temperatures.

B Plausible because RCP thermal barrier heat exchangers lose cooling water due to containment isolation of ICCW, but RCP seal injection remains in service.

C Correct answer. CRDs and RCP motor coolers lose cooling water due to containment isolation of ICCW and NSCCW.

D Plausible because RCP motor coolers and RCP thermal barrier heat exchangers lose cooling water due to containment isolation of NSCCW and ICCW, however RCP seal injection prevents high seal temperatures.

Comments None.

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AmerGen	TMI - Unit 1 Operating Procedure	Number 1105-3

Title	Revision No.
Safeguards Actuation System	**41**

Applicability/Scope	Responsible Office	Effective Date
TMI Division	Director, Operations	02/29/00

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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Originator	[illegible]	2/21/00
Procedure Owner	[illegible]	2/21/00
PRG	[illegible]	2/22/2000
Approver	[illegible]	2/22/2000

C. 30 PSIG R.B. Pressure Actuation

The following valves close automatically on 30 PSIG R.B. Pressure actuation.

A Actuation	A and B Actuation	B Actuation
I.C. System		
IC-V-3	IC-V-4 IC-V-6	IC-V-2
R.C. Pumps Seal Return		
MU-V-25		MU-V-26
R.C. Pump Motor Cooling Water		
NS-V-4	NS-V-15	NS-V-35

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GPU NUCLEAR

	TMI Emergency Procedure	Number **1202-8**
Title **CRD Equipment Failure**		Revision No. **53**
Applicability/Scope TMI Division	Responsible Office Plant Operations Director	Effective Date **10/12/99**

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator		10/7/99
Procedure Owner		10-8-99
PRG		10/9/99
Approver		10/8/99

E. **STATOR HIGH TEMPERATURE**

1E. Symptoms

1. Computer alarms and printouts of the affected mechanism(s) giving the stator temperature.

2. Possible low ICCW CRD cooling flow as indicated on IC10-FI or MAP alarm C-1-2.

3. Possible ICCW CRD filter dP High alarm on MAP C-1-4.

4. Possible ICCW cooler outlet temperature high on IC6-TI or MAP alarm C-2-3.

5. Possible CRD cooling water outlet temperature high on IC9-TI.

6. Possible IC system CRD cooling water outlet temperature high alarm on MAP C-1-3.

2E. Immediate Actions

1. Automatic Actions

a. MU-V-1A/B close when CRD cooling water outlet temperature reaches 160°F.

2. Manual Actions

NOTE
The steps with an asterisk (*) will be re-verified as the first step of the follow-up actions.

*a. Verify MU-V-1A/B close if CRD outlet temperature reaches 160°F.

*b. Check the ICCW system for proper operation by verifying

1. The following parameters for the CRD system:

i. Flow to CRD stators > 100 gpm.
ii. CRD return temperature < 160°F
iii. dP across IC-F-1A/B.

c. If required to provide continued cooling to the CRDMs, then start the standby IC-P-1 and/or valve in the standby IC-F-1.

3E. Follow-up Actions

Objective: The objective of this procedure is to prevent stator failure due to a loss of cooling water flow or high temperature in the ICCW system.

1. Reverify the steps in the Immediate Manual Actions that are marked with an asterisk (*). Use redundant indications where available.

CONTROLLED COPY

GPU NUCLEAR	TMI Abnormal Procedure	Number 1203-16

Title

Reactor Coolant Pump and Motor Malfunction

Revision No. **40**

Applicability/Scope	Responsible Office	Effective Date
TMI Division	Plant Operations Dir.	10/12/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

SIMULATOR ROOM FILE COPY

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	Signature	Date
Originator	R S Campbell FOR F. KALINKO	10-12-99
Procedure Owner	R S Campbell	10-12-99
PRG	[illegible]	10/12/99
Approver	James Powell	10/12/99

2. Increased monitoring of SLO flow is performed as specified by the Plant Operations Director.

3. Additional precautions are taken to insure that a standby makeup pump is available to provide seal injection in the case where the operating makeup pump would trip.

c. RC Drain Tank level control considerations may restrict continued operation with No. 2 Seal degradation. Limiting aspects of RC Drain Tank control include level, temperature and maintaining additional tank capacity to receive what must be rejected from the RC Drain Tank.

d. If No. 1 Seal leakoff is low and below the recommendations of OP 1103-6, it may be indicative of either an abnormally tight No. 1 seal or most likely an abnormally open No. 2 Seal. A bucket check of No. 1 seal leakoff per OP 1103-6 is recommended under these circumstances. Continued operation of the RCP within the guidance provided above can be expected.

2.0 LOSS OF RCP MOTOR COOLING WATER

2.1 Symptoms

1. Loss of N.S.

a. Motor air temp. Hi computer alarm Pts. A0699, A0705, A0711, A0717.

b. RCP stator temp. Hi computer alarm Pts. A0704, A0710, A0716, A0722.

c. Motor upper guide bearing temp. Hi computer alarm Pts. A0702, A0708, A0714, A0720. Motor lower guide bearing temp. Hi computer alarm Pts. A0703, A0709, A0715, A0721.

d. Motor up thrust bearing temp. Hi computer alarm Pts. A0701, A0707, A0713, A0719. Motor down thrust bearing temp. Hi computer alarm Pts. A0700, A0706, A0712, A0718.

2.2 Immediate Action

A. Automatic Action

None

B. Manual Action

1. On loss of N.S., if any of the following temperatures are exceeded, reduce power to 50-75 Percent and trip the affected pump.

a. Motor Stator > 150°C. (302°F).

Form ES-401-6

Written Exam Question Worksheet

Q # 044

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 015 K3.06 Page 3.7-5 Tier # 2

RO/SRO Importance Rating 2.9 3.2 Group # 1

Measurement Knowledge of the effect that a loss or malfunction of the NIS will have on the following: Reactor regulating system

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

Plant conditions:
- Reactor power is 60%, following response to an asymmetric (dropped) rod in Group 5.
- Reactor power channel SASS mismatch is present due to the dropped rod.
- All ICS stations are in AUTO.
- NI-5 is selected for ICS control.

With no operator action, from the list below identify the ONE (1) set of events that will initially result if NI-5 output fails to 0%.

A. Control rods will NOT insert/withdraw, AND feedwater flow will DECREASE.

B. Control rods will NOT insert/withdraw, AND feedwater flow will NOT change.

C. Control rods will WITHDRAW, AND feedwater flow will DECREASE.

D. Control rods will WITHDRAW, AND feedwater flow will INCREASE.

Technical Reference ARP MAP G-2-3, CRD Outmotion Inhibit, Rev. 6, Pages 1 and 2.
ARP MAP H-1-4, Neutron X-Limit to FW, Rev. 12, Page 1.

Open Exam Reference

Learning Objective IV.E.27.60 Given a set of plant conditions, determine the correct ICS response and any actions that must be taken if the ICS response is incorrect.

Question Source ☐ New ☑ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question # 46

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .6

Discriminant Validity Statements

A Plausible because of actual reactor power being at 60% (auto out inhibit), and neutron cross-limits will cause FW to decrease.

B Plausible because of actual reactor power being at 60% (auto out inhibit), and misconception that FW will also not change.

C Correct answer because indicated power (NI-5) will be less than 60% causing large positive neutron error, and neutron cross-limits will cause FW to decrease.

D Plausible because indicated power (NI-5) will be less than 60% causing large positive neutron error, and misconception that FW will change in same direction.

Comments None.

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GPU NUCLEAR	TMI Alarm Response Procedure	Number: MAP G
Title: **Main Annunciator Panel G**		Revision No.: **52**
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: **11/12/99**

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

TRAINING DEPT.

G-1-1 Rev. 12 11/12/99	G-1-2 Rev. 10 09/08/99	G-1-3 Rev. 5 09/08/99	G-1-4 Rev. 7 09/08/99	**G-1-5 Rev. 11 11/12/99**	G-1-6 Rev. 7 09/08/99	G-1-7 Rev. 8 09/08/99	G-1-8 Rev. 13 09/08/99
G-2-1 Rev. 11 09/08/99	G-2-2 Rev. 8 09/08/99	G-2-3 Rev. 6 09/08/99	G-2-4 Rev. 6 09/08/99	**G-2-5 Rev. 9 11/12/99**	G-2-6 Rev. 8 09/08/99	G-2-7 Rev. 10 09/08/99	G-2-8 Rev. 8 09/08/99
G-3-1 Rev. 6 09/08/99	G-3-2 Rev. 6 09/08/99	G-3-3 CANCELLED 05/06/92	G-3-4 Rev. 7 09/08/99	G-3-5 Rev. 7 09/08/99	G-3-6 Rev. 14 09/08/99	G-3-7 CANCELLED 10/15/91	G-3-8 Rev. 10 09/08/99

	Signature	Date
Originator	H L Nilson for E.G. Lawrence	10/20/99
Originator	Brantley for Dave Hass	10/30/99
Procedure Owner	Stuart Brantley	10/30/99
PRG	[illegible]	11/8/99
Approver	James N Paules	11/3/99

GPU NUCLEAR	TMI Alarm Response Procedure	Number MAP G
Title Main Annunciator Panel G		Revision No. (See Cover Page)

G-2-3
Revision 6

TRAINING DEPT.

ALARM:

CRD OUTMOTION INHIBIT

SET POINTS:

2 DPM Source Range & Reactor Power ≤ 10%
3 DPM Inter. Range
Asymmetric rod at ≥ 9" (From group avg. position) at power level ≥ 60%.

CAUSES:

Rods being withdrawn too long.
≥ 60% power with CRD's in auto and an asymmetric rod condition (≥ 9" from Group Average Position)
Loss of safety rod group out-limit signal
Dropped rod

AUTOMATIC ACTION:

Hi SUR on Console and RPS Cabinets
Asymmetric rod and if above 60% and CRD's in auto, a runback to 55% at 30%/min. will occur.

OBSERVATION (CONTROL ROOM):

Console CC--SUR meter indicates Hi
Rod position indication might indicate a dropped rod if absolute position indication is selected.
Safety rod "Out Limit" light off.

MANUAL ACTION REQUIRED:

If Hi SUR, drive rods in to decrease SUR.
If an asymmetric rod:
a. Verify power is reduced to < 60% of the power allowable for the RCP combination.
b. Determine which rod.

TRAINING DEPT.

MANUAL ACTION REQUIRED: (Cont'd)

CAUTION

If the ICS is calling for out motion while an OUT INHIBIT is present a large neutron error may develop. If the OUT INHIBIT clears as a result of pressing FAULT RESET or going below 60% reactor power, significant rod out motion may occur.

DO NOT PRESS FAULT RESET while a large neutron error is present if the CRD panel is in AUTO. Check neutron error periodically. If large neutron error develops take action to reduce neutron error, or place the CRD panel in MANUAL.

c. Refer to E.P. 1202-8.

CONTROLLED COPY

AmerGen	TMI - Unit 1 Alarm Response Procedure	Number **MAP H**
Title **Main Annunciator Panel H**		Revision No. **66**
Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date **03/23/00**

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

TRAINING DEPT.

H-1-1 **Rev. 20** **03/23/00**	**H-1-2** **Rev. 20** **03/23/00**	**H-1-3** **Rev. 9** **03/23/00**	**H-1-4** **Rev. 13** **03/23/00**	**H-1-5** **Rev. 12** **03/23/00**	**H-1-6** **Rev. 18** **03/23/00**	**H-1-7** **Rev. 18** **03/23/00**	**H-1-8** **Rev. 13** **03/23/00**
H-2-1 **Rev. 14** **03/23/00**	**H-2-2** **Rev. 14** **03/23/00**	**H-2-3** **Rev. 12** **03/23/00**	**H-2-4** **Rev. 10** **03/23/00**	**H-2-5** **Rev. 11** **03/23/00**	H-2-6 Cancelled 10/16/91	H-2-7 Cancelled 10/16/91	**H-2-8** **Rev. 9** **03/23/00**
H-3-1 Cancelled 10/16/91	**H-3-2** **Rev. 16** **03/23/00**	**H-3-3** **Rev. 13** **03/23/00**	**H-3-4** **Rev. 10** **03/23/00**	**H-3-5** **Rev. 8** **03/23/00**	H-3-6 Cancelled 10/16/91	**H-3-7** **Rev. 9** **03/23/00**	**H-3-8** **Rev. 10** **03/23/00**

	Signature	Date
Originator	George J Byler	03/03/2000
Originator		3/14/2000
Procedure Owner		3/15/2000
RG		3/16/2000
Approver		3-15-00

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TRAINING DEPT

ALARM:

NEUTRON X-LIMIT TO FW

SET POINTS:

Neutron Error of ± 5 percent.

CAUSES:

Neutron power changing faster or slower than neutron demand so that a neutron error of ± 5 percent is experienced.

1. Dropped Rod
2. Loss of controlling/selected NI.
3. Rx unable to respond fast enough to change in demand.

AUTOMATIC ACTION:

Neutron power 5 percent < neutron demand decreases F.W. demand.
Neutron power 5 percent > neutron demand increases F.W. demand.
Unit goes to track.

OBSERVATION (CONTROL ROOM):

Change in F.W. demand as seen on the "measured variable - Position" indicator on ICS "SG A(B) FW Demand" HAND/AUTO control stations.
Neutron error >± 5 percent.

MANUAL ACTION REQUIRED:

Verify auto action.

CAUTION

If taking manual action on either or both ICS "SG A(B) FW Demand HAND/AUTO control stations do so carefully coordinating with RCS parameters so as not to cause a Rx trip.

If either or both ICS "SG A(B) FW Demand" HAND/AUTO control stations are in HAND, place MV-Pos, switch to "MV" (reading input-output) use the "Raise - Lower" switch adjust to zero error (50 percent on MV-POS indicator).

Form ES-401-6

Written Exam Question Worksheet

Q # 045

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 015 K4.08 Page 3.7-6 Tier #

RO/SRO Importance Rating 3.4 3.7 Group #

Measurement Knowledge of NIS design features and/or interlocks provide for the following: Automatic rod motion on demand signals

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** D

Plant conditions:
- Reactor power is 85% following a failure in ICS SG/Rx Master.
- ICS stations in HAND:
  - SG/Rx Master.
  - Reactor Demand.
  - BOTH FW Loop Masters.
- All other stations, including the Diamond Rod Control Panel, are in AUTO.
- Control rod index is 275%.

The reactor operator inserts Group 8 rods from 30% to 20% to adjust core power imbalance. This action adds positive reactivity to the core and causes an increase in reactor power.

From the list below identify the ONE (1) statement that describes Control Rod Group 7 response during the insertion of Group 8 rods under the conditions described above.

A. Not move because neutron error will be zero (0).

B. Not move because Group-8 is being inserted.

C. Withdraw due to a positive ICS neutron error.

D. Insert due to a negative ICS neutron error.

Technical Reference OPM Section F-01, Control Rod Drive System, Rev. 7, Page 87.
OPM Section F-03, Integrated Control System, Rev. 9, Page 46.

Open Exam Reference

Learning Objective IV.E.27.60 Given a set of plant conditions, determine the correct ICS response and any actions that must be taken if the ICS response is incorrect.

Question Source ☐ New ☑ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question # 47

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☐ 55.45

Discriminant Validity Statements

A Plausible because Reactor Demand station is in HAND (but Diamond remains in AUTO).

B Plausible if misconception exists that Group 8 being selected on the diamond will prevent Group 7 auto motion.

C Plausible because Group-8 is being inserted.

D Correct answer.

Comments Question slightly modified to reflect reactivity change associated with Group 8 position. Does not meet significant modification definition.

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GPU NUCLEAR

TMI
Operating Procedure

Number: **1105-9**

Title: **Control Rod Drive System**

Revision No.: **56**

Applicability/Scope: TMI Division

Responsible Office: Plant Operations Dir.

Effective Date: **11/25/98**

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	Stuart Brantley	11-17-98
Procedure Owner	Stuart Brantley	11-17-98
PRG	[illegible]	11-19-98
Approver	L. G. [illegible]	11-18-98

b) IN Command

In order to get a MANUAL IN command the clamping relay K120 must be de-energized to allow the K120 contact in the command circuit to close. The INSERT/WITHDRAWAL switch on the Diamond Control Panel must be placed in the INSERT position. Group 8 command enable relays K8A and K8B must be de-energized and Diamond Control Panel must be in MANUAL. When these conditions are met relays K14A and K14B energize to close contacts in the motor control circuit.

❷ Automatic Commands for Groups 1 through 7

a) OUT MOTION

When the Diamond Control Panel is in AUTOMATIC the IN and OUT COMMANDS signals come from the REACTOR DEMAND Station of the ICS (refer to the ICS section of the plant manual). In order for the signal to energize the command relays for OUT MOTION (K15A and K15B) the clamping relay K120 must be de-energized. The AUTO OUT INHIBIT relay K33 must be energized, no IN command present (relays K14A and K14B de-energized) no SUR INHIBIT signal is present (relay K32 is energized) and the Diamond Control Panel must be in AUTOMATIC (relay K11B energized).

b) IN MOTION

In order to energize the In command relays K14A and K14B when Diamond is in AUTOMATIC (relay K11B energized) the clamping relay K120 must be de-energized, an IN command must be present from the ICS and AUTO INHIBIT relay K34 must be energized.

Any time an IN or OUT COMMAND signal energizes relays K15A, K15B, K14A, K14B (OUT and IN COMMAND relays) the OUT and IN MOTION lamps on the Diamond Control Panel energize. Since Group 8 has its own set of command relays the OUT and IN lamps will not be energized when moving Group 8 rods.

❸ OUT MOTION Command for Group 8

Group 8 rods can be moved for axial power shaping when Diamond is in AUTOMATIC or MANUAL. Group 8 must be selected on the GROUP SELECTOR switch to energize command enable relays K8A or K8B. The INSERT/WITHDRAWAL switch must be in the WITHDRAWAL position, no IN command present (relays K148A and K148B de-energized), and no SUR INHIBIT signal present (relay K32 energized). When all of these conditions are met, OUT command relays K158A and K158B will energize.

FOR INFORMATION ONLY

SECTION F-03
REVISION 9

TMI OPERATIONS PLANT MANUAL
INTEGRATED CONTROL SYSTEM

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Concur

Owner _____ Date 5/1/98

Operations Training Manager, TMI _____ Date 5/6/98

Training Coordinator, TMI _____ Date 5-7-98

Approved

Plant Operations Director TMI L. G. Noll Date 5-8-98

7.0 REACTOR CONTROL SUBSYSTEM

7.1 Purpose

The Reactor Control Subsystem is responsible for varying the heat output of the reactor. The subsystem must vary the total reactor heat output in such a manner so that the unit generation demand is satisfied. While satisfying this demand, the ratio between the heat output from the reactor and the heat removal capability of the feedwater must be maintained to hold the reactor average temperature at setpoint.

7.2 Rod Control

As discussed earlier, the amount of heat being generated by the reactor is a function of the position of the control rods in the reactor. Due to the fact that there are some other materials, uncontrollable poisons and soluble poisons such as boric acid, which can affect the total heat output from the reactor, the use of rod position as a measure of reactor power is not satisfactory. Through the use of out-of-core neutron detectors, which give a relative measure of the neutron flux, an accurate indication of reactor power can be obtained. Since the out-of-core neutron detectors measure a relative amount of energy, external to the core, the absolute reactor power is determined by a cycle heat balance. The neutron power measurement is calibrated to correspond to this heat balance information. When the heat balance indicates 100% thermal power, the neutron power measurement is adjusted to indicate 100%.

A demand (refer to Figure 18) referred to as "neutron power demand" is compared to the actual reactor neutron power, and the difference, neutron error, can be used to control the movement of the control rods. The function of the control rods is to maintain the neutron error as close to zero as possible. The Reactor Control Subsystem sends a signal to the rod control system to insert or withdraw the rods. In order to reduce control rod movement a deadband control is used. If the neutron error exceeds +1.1% (neutron power demand 1.1% greater than neutron power), a signal will be sent to the rod control system to withdraw the rods. The rods will continue to withdraw until the neutron error reduces below +.9%. The converse is true if a -1.1% neutron error exists; an "insert" signal will be initiated to the rod control system which will continue until the neutron error reduces below -.9%.

7.3 Reactor and Neutron Power Demand

The neutron demand can be established in one of two ways:

a. The operator can establish manually, through the use of the reactor control station, a neutron power demand for the control rods. In this mode of operation, any condition which will affect the neutron power level in the reactor (i.e., average temperature, boric acid concentration, xenon build-up) will be overcome by rod movement, holding the power level at the manually established value.

b. The second method of establishing a neutron power demand is with the reactor control station in automatic. In this mode of operation the neutron power demand becomes a function of reactor demand with a modification from T_{avg} control. The reactor demand is a function of the steam generator/reactor demand. The demand signal from the steam generator/reactor demand station is limited to a minimum of 15% in the Integrated Master Subsystem. This 15% minimum limit is placed on the demand because of the relationship of temperature with load (Figure 19).

Form ES-401-6

Written Exam Question Worksheet

Q # 046

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 004 A2.02 Page 3.1-18 Tier # 2

RO/SRO Importance Rating 3.9 4.2 Group # 1

Measurement Ability to predict the impacts of the following malfunctions or operations on the CVCS; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Loss of PZR level (failure mode)

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

The following plant conditions exist:
- a pressurizer temperature RTD develops an open circuit
- pressurizer differential pressure on the PPC (A0501) is 300"
- RCS pressure is 2155 psig and stable

Which ONE (1) of the following identifies the manner in which the temperature instrument fails AND what the actual pressurizer level is?

A. low, 120"

B. low, 208"

C. high, 120"

D. high, 208"

Technical Reference OPM Section N-11, Instrumentation and Control, Rev. 2 page 56
EP 1202-29, Pressurizer System Failures, Rev. 54 pages 12 and 15

Open Exam Reference 1202-29, Pressurizer System Failures, Rev. 54 page 15

Learning Objective IV.E.01.02/ V.D.11.06 Describe the failure modes of the following instruments:
a. RTD
Determine actual PZR level by converting the detector D/P output to level using Figure 1 of EP 1202-29.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☑ 55.43 .5 ☑ 55.45 .3, .5

Discriminant Validity Statements

A Plausible, wrong failure direction.

B Plausible, but swaps axis on the graph

C Correct answer.

D Plausible, wrong failure direction and swaps axis on the graph

Comments None.

TMI-1 OPERATIONS PLANT MANUAL
INSTRUMENTATION AND CONTROL

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Concur

Author N/A Date

Owner Date 7/6/93

Operator Training Manager, TMI-1 Date 7/12/93

Training Coordinator, TMI-1 Date 7/14/93

Approved

Plant Operations Director, TMI-1 Date 7/20/93

Rosemount 414L temperature monitoring equipment is used extensively at TMI Unit 1. A simplified schematic of the 414L temperature monitoring circuit is shown in Figure 23 A. This circuit is more complex, but similar in operation to the 4-wire RTD circuit shown in Figure 22. It uses a constant current power supply to feed the temperature monitoring circuit. The dashed lines around the circuit represents the power supply's and monitoring circuit's enclosure. Connections to the enclosure are indicated by circled numbers. The power supply's connections to the monitoring circuit are shown with + and - signs. The circuit's output is sensed by a high-impedance voltmeter connected between resistors F and E (point 4), and resistors J and RTD (point 3). Figure 23 B shows a straight line schematic of the circuit. Resistors Z, J, F, E, and the RTD form two series legs in a parallel network. The millivolt output is sensed by the voltmeter that bridges this network.

The constant current from the power supply splits at resistor Z. Part of the current goes through the upper series leg of resistors Z, F, and E, and the parallel resistors P and N. The rest of the current goes through the lower series leg of resistors Z, J, and RTD, and the parallel resistors P and N. The circuit's Mv output is the difference in voltage between points 3 and 4. This voltage difference is proportional to the temperature sensed by the RTD.

When the temperature sensed by the RTD increases, the RTD's resistance increases. This causes the current through the lower series circuit and the RTD to decrease. This results in a decreased voltage felt at point 3. Since the power supply is putting out a constant current, the current through the upper series circuit increases. This increases the voltage felt at point 4. The resultant increase in the voltage applied to the output circuit is directly proportional to the temperature sensed at the RTD. The voltmeter measures this voltage and is calibrated to indicate the temperature at the RTD. A sensed temperature decrease has the opposite effect.

Temperature compensation is accomplished in the same manner as previously discussed.

Loss of power to an RTD circuit causes the indication to fail to 0 percent. Since an RTD's resistance increases with temperature, an open RTD will fail the indication high because an open circuit gives infinite resistance. Conversely, a shorted RTD represents zero resistance and will fail the indication low.

4. Thermocouples

Thermocouples are used extensively throughout most power plants due to their simplicity and reliability. They are seldom direct indicating devices. They are usually connected to multipoint recorders or computers. Thermocouples are used to detect incore temperatures as well as temperature throughout the secondary plant.

FOR INFORMATION ONLY

GPU NUCLEAR	TMI Emergency Procedure	Number 1202-29
Title: **Pressurizer System Failure**		Revision No. 54
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: 07/06/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	[illegible]	6-21-99
Procedure Owner	[illegible]	6-24-99
PRG	[illegible]	6/29/99
Approver	James N Faulks	6/25/99

4. Verify heaters re-energize if loss was due to 80 inch interlock or attempt to re-energize heaters and regain pressure control.

> **NOTE**
>
> Use the lowest indication if greater than 80 inches for heater protection/operation and the highest to prevent filling the system solid.

D.3 Follow-Up Action

Objective:

Select an operable instrument and return system parameters to normal or shutdown plant while maintaining RCS inventory if all pressurizer level is lost.

_____ 1. If pressurizer level recorder indication is lost, select another transmitter or use the computer for level indication. Verify backup PRZ level indication console cc (LI777A). (Temp Comp. by RC2-TE2).

a. If computer level indication is selected, be sure to select temperature compensated pressurizer level (Pt. C1720 and C1722).

b. Label failed instrument channel out of service and initiate repair.

> **NOTE**
>
> If temperature compensated pressurizer level is not available, d/p may be used but must be compensated using Figure 1. (Computer points AO501 and AO503). Uncompensated level may be read at the ICS/NNI cabinets on RC1-LI1 and RC1-LI2 but must be compensated using Figure 2.

> **NOTE**
>
> Comply with Tech Spec 3.5.5 which requires a shutdown to commence within 48 hours after decreasing below the minimum number of channels (1) required to be operable.

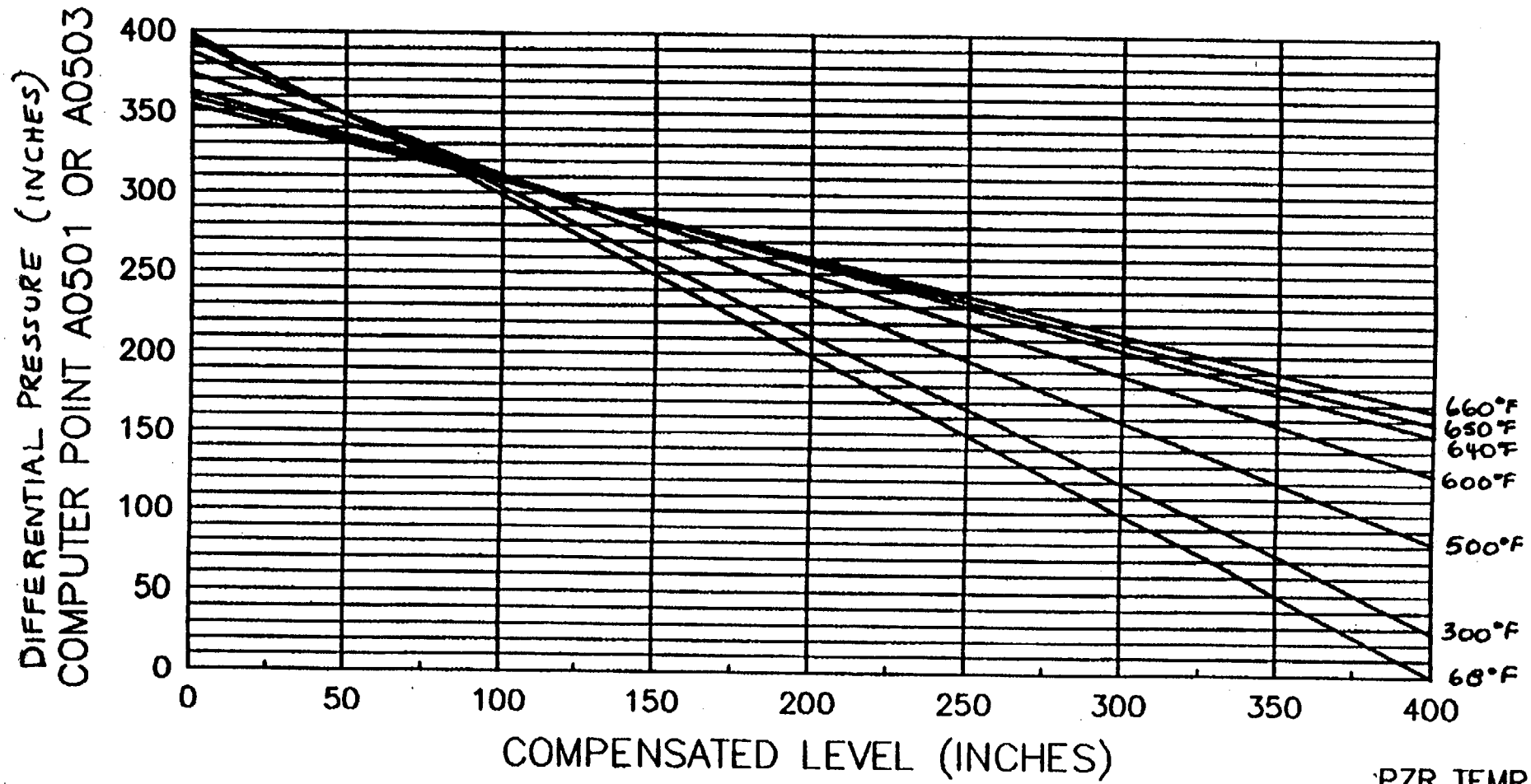
_____ 2. If all control room indications of pressurizer level are declared inoperable, send C.R.O. to the remote shutdown panel to read pressurizer level and with communications from the panel to the control room, commence a plant shutdown at the direction of the Plant Operations Director.

_____ 3. If all available pressurizer level instrumentation is lost, commence a Reactor Shutdown to comply with Tech. Spec. 3.5.5.

a. This occurrence assumes that pressurizer level was in its normal range before the failure of all the level instrumentation. Figure 3 or Table 1 should be used to determine the amount of water to be added to the RCS during shutdown to minimize changes to pressurizer level. Abnormal Transient Rules, Guides and Graphs, 1210-10, Figure 2 would be helpful for known cooldown rates for a makeup flow rate.

_____ 4. Refer to reporting procedure Administrative Procedure 1044 and make appropriate notification.

FIGURE 1
Differential Pressure vs Compensated Level at Various Pressurizer Temperatures



SOURCE: B&W/BAILEY MODULE CALIBRATION DOCUMENT, PAGES RC7,RC9,RC11,RC13, AND RC14

PZR TEMP
CONSOLE CTR IND OR
COMP PT #A0504

FOR INFORMATION ONLY

GPU NUCLEAR	TMI Emergency Procedure	Number 1202-29
ıde		Revision No. 54
Pressurizer System Failure		
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: 07/06/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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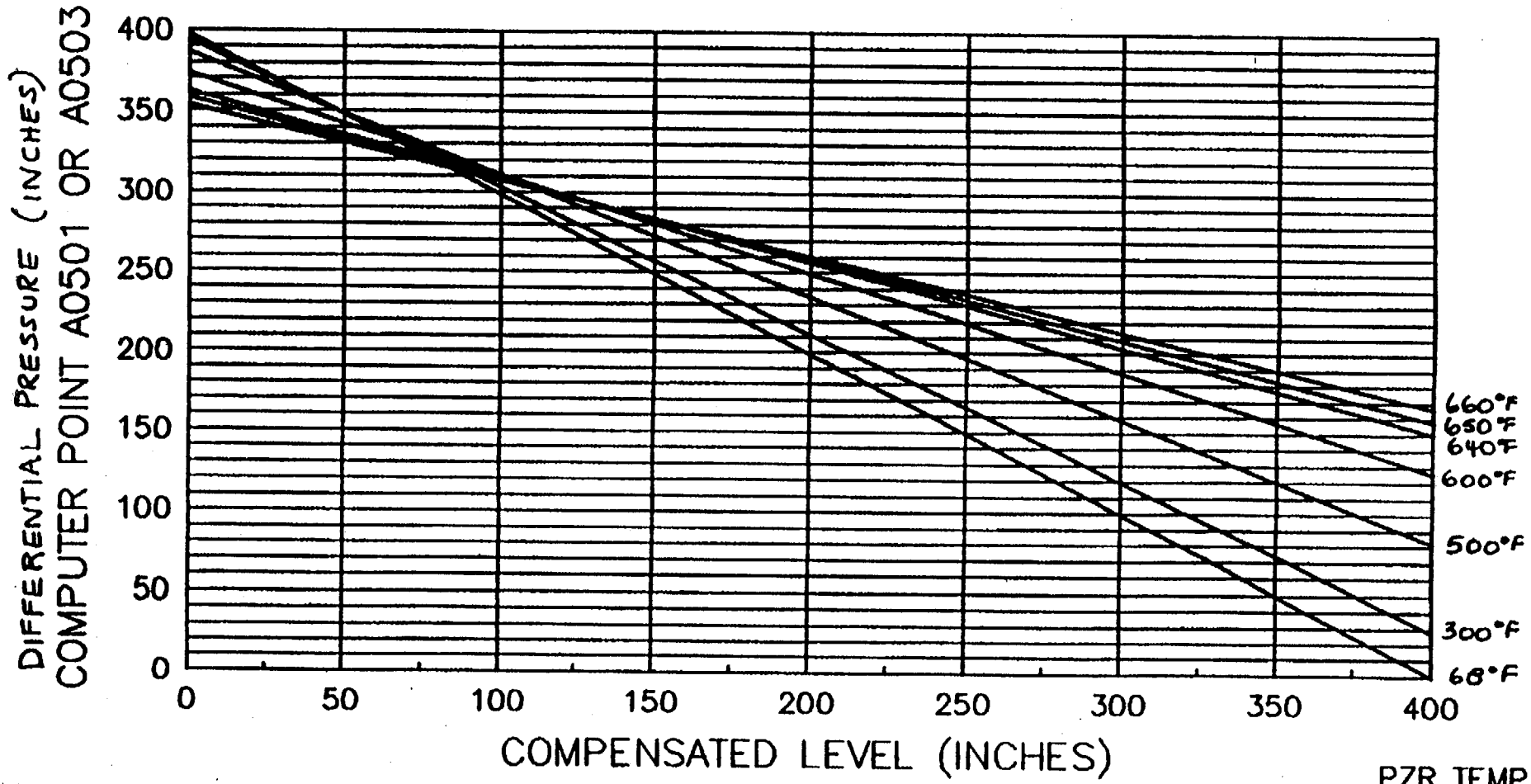
	Signature	Date
Originator		6-21-99
Procedure Owner		6-24-99
PRG		6/29/99
ıprover	James N Caulks	6/25/99

FIGURE 1
Differential Pressure vs Compensated Level
at Various Pressurizer Temperatures



SOURCE: B&W/BAILEY MODULE CALIBRATION DOCUMENT,
PAGES RC7,RC9,RC11,RC13, AND RC14

PZR TEMP
CONSOLE CTR IND OR
COMP PT #A0504

Form ES-401-6

Written Exam Question Worksheet

Q # 047

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 017 A4.02 Page 3.7-12 Tier # 2

RO/SRO Importance Rating 3.8 4.1 Group # 1

Measurement Ability to manually operate and/or monitor in the control room: Temperature values used to determine RCP operation during inadequate core cooling

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

Which ONE (1) statement does NOT satisfy the Superheat Determination/Limit Rule as defined in ATP-1210-10?

A. 25° F of superheat as determined by the plant computer.

B. 25° F of superheat as determined by the most conservative of the two subcooling margin meters on panel PCL.

C. 25° F of superheat as determined by the average of 5 highest operable incore thermocouples and RCS wide range pressure.

D. 25° F of superheat as determined by the highest operable BIRO incore thermocouple and RCS wide range pressure.

Technical Reference ATOG 1210-10 Abnormal Transient Rules, Guides and Graphs, Rev. 34 page 6.

Open Exam Reference

Learning Objective V.E.10.01 Without the use of ATP 1210-10 Abnormal Transient Rules, Guides and Graphs, be able to recite and explain the rules and the basis of the rules contained in the procedure.

Question Source ☐ New ☑ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☑ Used in Last Two TMI NRC Exams Exam Date 1998 SRO EXAM

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .5, .8

Discriminant Validity Statements

A This answer meets requirements for one of three methods to determine existence of superheat in the RCS IAW ATP 1210-10.

B Correct answer. This answer does not meet requirements of ATP 1210-10.

C This answer meets requirements for one of three methods to determine existence of superheat in the RCS IAW ATP 1210-10.

D This answer meets requirements for one of three methods to determine existence of superheat in the RCS IAW ATP 1210-10.

Comments None.

CONTROLLED COPY

GPU NUCLEAR

	TMI Abnormal Transient Procedure	Number **1210-10**
Title **Abnormal Transients Rules, Guides and Graphs**		Revision No. **37**
Applicability/Scope TMI Division	Responsible Office Plant Operations Dir.	Effective Date 10/15/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

SIMULATOR ROOM FILE COPY

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	Signature	Date
Originator	[illegible]	10-14-99
Procedure Owner	James P Paulles	10/15/99
PRG	[illegible]	10/15/99
Approver	James P Paulles	10/15/99

1.6.1 If 25°F subcooling margin is lost raise level in the operable OTSG(s) to 75-85% (Operating Range).

1.6.2 Maintain OTSG level ≥ 25" start-up range with RCPs on.

1.6.3 Maintain OTSG level ≥ 50% operate range with RCPs off.

1.7 Superheat Determination/Limit - Rule

1.7.1 Superheat exists at 25°F of superheat as determined by the value of Comp. Pt. C4132 (calculated Incore Superheat).

1.7.2 Superheat exists at 25°F of superheat as determined by the average of the five (5) highest, operable incore thermocouples (Comp. Pt. C4006) and RCS wide range pressure PI-949A (as read on panel PCL) or PT-963 (Comp. Pt.A0404).

1.7.3 Superheat exists at 25°F of superheat as determined by the highest operable BIRO incore thermocouple and RCS wide range pressure PI-949A (as read on panel PCL) or PT-963 (Comp. Pt. A0404).

2.0 GUIDELINES

2.1 Pressurizer Level Guide

If the reactor power is > 25% and pressurizer level cannot be maintained ≥ 150 inches, then trip the reactor and initiate HPI.

2.2 Emergency Boration Guide

Definition: Emergency Boration is the addition of boric acid to the RCS until 1% dk/k shutdown has been achieved for the expected plant condition as determined by Figure 1 of OP 1103-4, "Soluable Poison Concentration Control", or by OP 1103-15A, "Reactivity Balance".

2.2.1 Emergency Boration may be accomplished from any one of the following sources: (Refer to OP 1103-4 as necessary)

a. BWST

b. BAMT

c. RBAT

2.2.2 Emergency Borate using BWST

a. Open MU-V-14A or B

b. Establish a RCS letdown and bleed path if required to achieve the boration objective defined above.

Form ES-401-6

Written Exam Question Worksheet

Q # 048

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 022 K1.01 Page 3.5-5 Tier # 2

RO/SRO Importance Rating 3.5 3.7 Group # 1

Measurement Knowledge of the physical connections and/or cause-effect relationships between the CCS and the following systems: SWS

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

Which ONE (1) of the following is the purpose of RR-V-6 throttling to maintain 60 psig back pressure when Reactor Building Emergency Cooling is in service?

A. Ensure the RB coolers are full of water at their highest elevation.

B. Ensure the RB coiling coils do not experience flow-induced vibration.

C. Prevent runout of a single Reactor River pump if the second pump fails to start.

D. Prevent radioactivity from reaching the river due to leakage in the RB cooling coils.

Technical Reference OPM Section D-04, Reactor Building Emergency Cooling River Water, Rev. 12 page 9

Open Exam Reference

Learning Objective IV.A.17.03 Describe the location, purpose, and any interlocks of the following components of the RB Emergency Cooling River Water:
3. RR-V-6

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .2 TO .9 ☐ 55.43 ☑ 55.45 .7 TO .8

Discriminant Validity Statements

A Plausible, used in other systems.

B Plausible, used in other systems.

C Plausible, could have this effect.

D Correct answer.

Comments None.

FOR INFORMATION ONLY

TMI OPERATIONS PLANT MANUAL

REACTOR BUILDING EMERGENCY COOLING RIVER WATER

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Concur

Owner L.G. Noll FOR R.E. BOYER Date 11-19-98

Operations Training Manager, TMI [signature] Date 11/24/98

Training Coordinator, TMI Linda L. Ritter Date 12-1-98

Approved

Plant Operations Director TMI L.G. Noll Date 12-1-98

c. Nuclear Services Closed Cooling Water

❶ Downstream of RR-V-8B is a crosstie to the Nuclear Services Closed Cooling Water System consisting of a 4 inch crosstie that is fitted with a 1/2 inch bypass (see Figure 1). The cross connect serves two purposes as follows:

a) The 4 inch line provides a means of flushing out the Reactor Building Emergency Cooling Coils after the initial functional test with river water and for providing a non-corrosive environment for the cooling coils.

b) The 1/2 inch line provides a means of detecting small leaks in the Reactor Building Emergency Cooling Coils. During normal plant operation, the coils are filled with demineralized water from the Nuclear Services Closed Cooling Water System. The combined fill and flushing valve, NS-V-8, is closed and the inlet and outlet valves for the rotometer are open. Should a small leak occur, flow through rotometer FE-76 will be indicated locally at the rotometer. The rotometer is capable of measuring flows from 0-5 gpm.

d. Remote Shutdown Panels

❶ For Remote Shutdown Sequences, the Auxiliary Remote Shutdown Panel "B" provides operation of only RR-V-1B from the Remote Shutdown Panels. RR-P-1B may be operated locally at the 1E 4160V Bus by using the emergency position on the 69 Switch Key.

❷ Refer to EP 1202-37 for sequence and operation of Reactor River Components during times when the Control Room/Relay Room is uninhabitable.

6.1.2 Automatic Pressure Control

The pressure in the Reactor Building Emergency Cooling Coil during emergency operation will be maintained at 60 psig at the outlet piping to ensure that the pressure in the Emergency Cooling Coils is higher than the pressure in the Reactor Building. This will ensure that any leaks in the coils will cause water to flow into the Reactor Building rather than allowing potentially radioactive vapor to pass into the Emergency Cooling Coils and from there to the river.

Form ES-401-6

Written Exam Question Worksheet

Q # 049

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 022 K3.01 Page 3.5-5 Tier # 2

RO/SRO Importance Rating 2.9 3.2 Group # 1

Measurement 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 ☑ SRO ☐ PRA Related **Correct Answer** C

A large steam leak has occurred, RB pressure is 15 psig.

From the list below, identify the ONE (1) indication in the Control Room that would indicate higher than the actual condition.

A. SCM

B. RB Sump Level

C. Pressurizer Level

D. RCS Pressure

Technical Reference ATOG 1210-10, Abnormal Transient Rules, Guides and Graphs, page 9 Rev. 37.

Open Exam Reference

Learning Objective IV.E.01.06 Describe the effects of Reactor Building temperature and pressure changes on process instrumentation.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge

☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .6

Discriminant Validity Statements

A Plausible if misunderstanding of temperature compensation feature is present.

B Plausible if heating/pressure effects on sump level indication is misunderstood.

C Correct answer.

D Plausible if atmospheric effects on pressure cell are misunderstood.

Comments None.

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Q-49

GPU NUCLEAR

TMI
Abnormal Transient Procedure

Number: 1210-10

Title: **Abnormal Transients Rules, Guides and Graphs**

Revision No.: 37

Applicability/Scope: TMI Division

Responsible Office: Plant Operations Dir.

Effective Date: 10/15/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

TRAINING DEPT.

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	Signature	Date
Originator	[illegible]	10-14-99
Procedure Owner	James R. Paules	10/15/99
PRG	[illegible]	10/15/99
Approver	James R. Paules	10/15/99

____		b.	IF	primary-to-secondary heat transfer does not exist;	THEN	go to ATP 1210-9.
____		c.	IF	primary-to-secondary heat transfer exists;	THEN	go to ATP 1210-9 Follow-up Actions.
☐	2.4.13	WHEN		neither OTSG is available as a **HEAT SINK**;	THEN	refer to ATP 1210-4 for guidance on recovering feedwater and establishing primary-to-secondary heat transfer.
☐	2.4.14	WHEN		**RCS PRESSURE** reaches 600 psig;	THEN	verify CFT discharge.
☐	2.4.15	WHEN		**LPI** > 1000 gpm per leg;	THEN	adjust feeding and steaming rates to control cooldown rate within T.S. limits AND go to ATP 1210-7.

2.5 Instrumentation Effects

2.5.1 Elevated Reactor Building (RB) Temperature/Rapid RCS Depressurization Effects on Instrumentation

Elevated Reactor Building temperature can affect fluid density in instrument reference legs and cause pressurizer and OTSG level indications to be erroneously high.

Instrument reference legs may be locally heated above average containment temperature. DO NOT rely solely on single level readouts.

Rapid RCS depressurization (to 600 psig within several minutes) will cause pressurizer level indication to be erroneously high due to off gassing and water ejection from the reference leg.

Extreme depressurization coupled with high reference leg temperature can cause reference leg boiling.

2.5.2 Low Flow Signal Cutouts

The instrument flow loops for HPI, EFW and LPI are subject to electronic cutouts at low flow conditions.

HPI flow of ≤ 43.3 GPM displays 0 on FI 1126 through 1129.

LPI flow of ≤ 433.3 GPM displays 0 on FI 802A and FI 803A.

EFW flow of ≤ 60.6 GPM displays 0 on FI 788, 789, 782, 791.

Form ES-401-6

Written Exam Question Worksheet

Q # 050

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 056 K1.03 Page Tier # 2

RO/SRO Importance Rating 2.6 2.6 Group # 1

Measurement Knowledge of the physical connections and/or cause-effect relationships between the Condensate System and the following systems: MFW

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

The following plant conditions exist:
- Reactor power 100%
- Condensate Booster pumps CO-P-2A and CO-P-2C are operating
- Feed Reg Valve demands are 52%

A shaft shear occurs on CO-P-2A.

Predict which ONE (1) of the following will be the INITIAL response to this condition.

A. CO-P-2B starts automatically.

B. FW to Rx Cross Limit will occur.

C. Feed Reg Valve demands will decrease.

D. Feed Pump discharge pressures increase.

Technical Reference ARP Main Annunciator Panel H, Rev 66 H-1-5.

Open Exam Reference

Learning Objective IV.E.27.39 With respect to cross limits:
d. Given a specific plant condition, predict and describe the effect of a cross limit
on the feedwater and/or Rx subsystem demand signals.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .2 TO .9 ☐ 55.43 ☑ 55.45 .7, .8

Discriminant Validity Statements

A Plausible since pump counter circuit will look for loss of condensate booster pump.

B Correct answer.

C Plausible since FRV will respond to this situation.

D Plausible since feed pump speed will vary under this condition.

Comments None.

AmerGen	TMI - Unit 1 Alarm Response Procedure	Number Q-050 **MAP H**

Title		Revision No.
Main Annunciator Panel H		**66**
Applicability/Scope	Responsible Office	Effective Date
TMI Division	Director, Operations	**03/23/00**

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

H-1-1 **Rev. 20** **03/23/00**	**H-1-2** **Rev. 20** **03/23/00**	**H-1-3** **Rev. 9** **03/23/00**	**H-1-4** **Rev. 13** **03/23/00**	H-1-5 Rev. 12 03/23/00	**H-1-6** **Rev. 18** **03/23/00**	**H-1-7** **Rev. 18** **03/23/00**	**H-1-8** **Rev. 13** **03/23/00**
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	Signature	Date
Originator	George J Butler	03/03/2000
Originator	[illegible]	3/14/2000
Procedure Owner	[illegible]	3/15/2000
:G	[illegible]	3/16/2000
Approver	[illegible]	3-15-00

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Title

Main Annunciator Panel H

H-1-5
Revision 12

TRAINING DEPT.

ALARM:

FW X-LIMIT TO RX

SET POINTS:

F W demand 5 percent greater than F.W. Flow.

CAUSES:

Loss of main F.W. Pump
S.G. on a BTU limit (Limiting FW Flow)
S.G. on a high level limit (Limiting FW Flow).

AUTOMATIC ACTION:

Reactor demand decreases causes inward rod motion.

OBSERVATION (CONTROL ROOM):

Reactor demand as indicated by MV-Pos indicator (in POS on ICS Reactor Demand HAND/AUTO control station) decreases.
Decrease in reactor Power as indicated by NI-5, 6, 7 and 8 on CC.

MANUAL ACTION REQUIRED:

Verify rods move in and Rx power decreases.

CAUTION

If operating ICS Reactor Demand HAND/AUTO control station in HAND raise or lower Rx power carefully coordinating with secondary side parameters so as not to cause a Rx Trip.

If ICS Reactor Demand HAND/AUTO control station is in HAND, use the raise-lower switch and decrease reactor power to clear the cross limit.
Verify alarm using redundant instrument, may have to read on computer.

Form ES-401-6

Written Exam Question Worksheet

Q # 051

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 056 A2.04 Page 3.4-39 Tier # 2

RO/SRO Importance Rating 2.6 2.8 Group # 1

Measurement Ability to predict the impacts of the following malfunctions or operations on the Condensate System; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Loss of condensate pumps

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

Initial plant conditions:
- Reactor power is 100%.
- CO-P-1A and CO-P-1B are running, CO-P-1C is in Normal-After-Stop.
- CO-P-2A and CO-P-2B are running, CO-P-2C is in Normal-After-Stop.
- CO-P-1B breaker trips on an electrical fault.
- After a period of two (2) seconds, CO-P-1C automatically starts.

From the list below, identify the ONE (1) statement that describes the plant response for these conditions.

A. One condensate booster pump will trip, both main feed pumps remain running, with an ICS runback.

B. One condensate booster pump and one main feed pump will trip, with an ICS runback.

C. One main feed pump will trip, both condensate booster pumps remain running, with an ICS runback.

D. Both main feed pumps trip, one condensate booster pump trips and the reactor trips.

Technical Reference OPM Section G-01, Condensate System, Rev. 13 page 7

Open Exam Reference

Learning Objective IV.C.02.10 State the interlocks associated with the condensate and condensate booster pumps.

Question Source ☐ New ☑ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question # 51

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☑ 55.43 .5 ☑ 55.45 .3, .13

Discriminant Validity Statements

A Plausible misconception regarding extremely complicated condensate-condensate booster pump counting circuit.

B Correct answer.

C Plausible misconception regarding extremely complicated condensate-condensate booster pump counting circuit.

D Plausible misconception regarding extremely complicated condensate-condensate booster pump counting circuit.

Comments None.

FOR INFORMATION ONLY

SECTION G-01
REVISION 13

TMI OPERATIONS PLANT MANUAL
CONDENSATE SYSTEM

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Concur

Owner — Randy S. Campbell — Date 2-4-99

Operations Training Manager, TMI — [illegible] — Date 2/11/99

Training Coordinator, TMI — Linda L. Ritter — Date 2-11-99

Approved

Plant Operations Director TMI — L. G. [illegible] — Date 2-12-99

Because the Condensate, Condensate Booster and Main Feedwater Pumps are connected in series, protection is provided to ensure adequate suction pressure to all pumps. A counting circuit ensures that an adequate number of pumps are running upstream before certain pumps may be started. For example, the second Booster Pump may not be started unless there are at least two Condensate Pumps operating. Similarly, the second Feedwater Pump may not be started unless there are at least two Booster Pumps operating. In the event a Condensate Pump trips and the standby pump fails to start within 0.5 seconds, one Booster Pump and one Feedwater Pump will be tripped. If the station is above 60% power, the Integrated Control System (ICS) will automatically reduce the load to a MWe load equivalent of 60% power. In the event a Booster Pump trips and the standby pump fails to start within 0.5 seconds, one Feedwater Pump will be tripped and the unit will run back as stated above. The last feed pump that was reset will trip.

5.1.3 Emergency Operation

In the event that Condensate Storage Tanks low Level Alarms occur as a result of the Condensate Storage Tanks being used to supply emergency feedwater, the operator may align the Hotwell directly to the Emergency Feedwater Pump suction by breaking vacuum, then positioning the "NORMAL - EMERGENCY" Hotwell Alignment Switch located on console center to the "EMERGENCY" position. This switch de-energizes three solenoid valves adjacent to CO-V-6, 7 and 8, which causes them to fail in the following manner:

CO-V-6	Shuts - prevents air from entering the Emergency Feed Pump suction.
CO-V-7	Shuts - prevents air from entering the Emergency Feed Pump suction.
CO-V-8	Open Lines up Emergency Feed Pump to the Hotwell.

If a 4160V Bus that is powering a string of CO-P-1's, CO-P-2's is lost, the ext. controls should be matched on the tripped pumps prior to starting the standby Condensate Pump. This will prevent auto starting of the standby Condensate Booster Pump.

5.2 Components

5.2.1 Condenser

The Condenser's major function is to condense the exhausted steam from the Low Pressure Turbines. A detailed description of the Condenser is contained in Section H-4 of this manual.

5.2.2 Condensate Pumps

There are three Condensate Pumps which are located on the east side of the Condenser. Each pump is capable of supplying 50% of the required Condensate flow at full power. The pumps take a suction from the Condenser Hotwell and discharge the Condensate at approximately 175 psi.

The Condensate Pumps are powered from:

CO-P-1A	TP 4160V Bus 1A
CO-P-1B	TP 4160V Bus 1B
CO-P-1C	TP 4160V Bus 1C

Form ES-401-6

Written Exam Question Worksheet

Q # 052

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 059 K3.03 Page 3.4-41 Tier # 2

RO/SRO Importance Rating 3.5 3.7 Group # 1

Measurement Knowledge of the effect that a loss or malfunction of the MFW will have on the following: S/Gs

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

The following plant conditions exist:
- A loss of condenser vacuum has occurred.
- Both MFPs are tripped.
- EFW actuated
- OTSG pressures are 1005 psig and stable.
- The IMAs of 1210-1 have been completed.

Which ONE (1) of the following describes the status of the OTSGs?

Tensile stress will be:

A. higher due to steaming through the ADVs.

B. higher due to the loss of MFW.

C. lower due to steaming through the ADVs.

D. lower due to the loss of MFW.

Technical Reference ATOG 1210-10, Abnormal Transien Rules, Guides and Graphs, page 15, Rev. 37.

Open Exam Reference

Learning Objective V.E.09.06 Given plant conditions and OTSG tube-to-shell Delta T determine what actions, if any, should be taken to limit/control OTSG tube-to-shell Delta T in accordance with ATP 1210-10, Abnormal Transient Rules, Guides, and Graphs.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .6

Discriminant Validity Statements

A Plausible since steaming rate affects delta T, but the steaming rate would be no different using TBVs.

B Correct answer.

C Plausible if evaluation is incorrect and assumption exists that delta T is affected by method of OTSG pressure conrol.

D Plausible if evaluation is incorrect.

Comments None.

FOR INFORMATION ONLY

GPU NUCLEAR	TMI Abnormal Transient Procedure	Number Q-052 1210-10
Title: **Abnormal Transients Rules, Guides and Graphs**		Revision No. 37
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: 10/15/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	[illegible]	10-14-99
Procedure Owner	James R. Faulks	10/15/99
PRG	[illegible]	10/15/99
Approver	James R. Faulks	10/15/99

2.12 OTSG Tube-to-Shell Delta-T Limit/Control

2.12.1 The tube-to-shell Delta-T values can be determined by Comp. Pt's C4015 and C4016.

2.12.2 IF the tensile limit (-70°F) is approached (tubes colder than shell); THEN perform the following actions:

a. Reduce the cooldown rate.

b. Feed the affected OTSG with MFW to cool the shell.

2.12.3 IF the compressive limit (+60°F) is approached (tubes hotter than shell); THEN perform the following actions:

a. Increase the cooldown rate.

b. Feed the affected OTSG with EFW to cool the tubes.

2.13 Pressurized Thermal Shock (PTS) Guidance

Definition: Pressurized Thermal Shock (PTS) is a rapid cooling of the reactor vessel, primarily the downcomer region, where the RCS pressure either remains high or rapidly increases while the vessel temperature is low.

Scope: The guidance applies to MU/HPI operation (not LPI) and should be followed until an engineering evaluation of the transient has been competed.

2.13.1 IF cooldown rate exceeds the Technical Specification limits; THEN reduce the cooldown rate to within T.S. limits and prevent violating RV P-T Limits.

2.13.2 IF RCPs are off AND HPI is on; THEN minimize subcooling margin.

2.14 Reactor Coolant Pump (RCP) Restart Criteria

2.14.1 RCPs may be restarted if the minimum emergency NPSH is available (See NPSH curve on Figure 1 or 1A) AND ≥ 25°F subcooling margin has been maintained throughout the transient.

Form ES-401-6

Written Exam Question Worksheet

Q # 053

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 059 A1.03 Page 3.4-43 Tier # 2

RO/SRO Importance Rating 2.7 2.9 Group # 1

Measurement Ability to predict/monitor changes in parameters associated with operating the MFW controls including: Power level restrictions for operation of MFW pumps and valves

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

A plant power reduction is being performed in order to secure the "B" Main Feedwater Pump due to high vibration.

Which ONE (1) of the following conditions would represent the HIGHEST power level acceptable for securing FW-P-1A WITHOUT exceeding an ICS High Load Limit?

A. Unit Load Demand at 65%

B. Unit Load Demand at 45%

C. Main Feedwater flow at 7.8 Mlbm/hr

D. Main Feedwater flow at 6.2 Mlbm/hr

Technical Reference ARP MAP H-1-1, ICS Runback, Rev. 20 page 1

Open Exam Reference

Learning Objective IV.E.27.60 Given a set of plant conditions, determine the correct ICS response and any actions that must be taken if the ICS response is incorrect.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge

☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☐ 55.43 ☑ 55.45 .5

Discriminant Validity Statements

A Plausible, may be mistaken to equate to reactor power being below the approximate 60% equivalent of the 585 Mwe ICS High Load Limit.

B Plausible, equates to 450 Mwe which is below the 585 Mwe ICS High Load Limit.

C Plausible because one of the limits for single MFW pump operation is 8 Mlbm/hr. However this equates to approximately 628 Mwe which is above the 585 Mwe ICS High Load Limit.

D Correct answer. Equates to approximately 500 Mwe which is below the 585 Mwe ICS High Load Limit.

Comments None.

AmerGen	TMI - Unit 1 Alarm Response Procedure	Number **MAP H**
Title **Main Annunciator Panel H**		Revision No. **66**
Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date **03/23/00**

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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H-2-1 **Rev. 14** **03/23/00**	**H-2-2** **Rev. 14** **03/23/00**	**H-2-3** **Rev. 12** **03/23/00**	H-2-4 Rev. 10 03/23/00	H-2-5 Rev. 11 03/23/00	H-2-6 Cancelled 10/16/91	H-2-7 Cancelled 10/16/91	**H-2-8** **Rev. 9** **03/23/00**
H-3-1 Cancelled 10/16/91	**H-3-2** **Rev. 16** **03/23/00**	**H-3-3** **Rev. 13** **03/23/00**	H-3-4 Rev. 10 03/23/00	H-3-5 Rev. 8 03/23/00	H-3-6 Cancelled 10/16/91	**H-3-7** **Rev. 9** **03/23/00**	**H-3-8** **Rev. 10** **03/23/00**

	Signature	Date
Originator	George J Butler	03/03/2000
Originator		3/14/2000
Procedure Owner		3/15/2000
G		3/16/2000
Approver		3-15-00

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TRAINING DEPT

ALARM:

ICS RUNBACK

SET POINTS:

Loss of one RCP and > 75 percent power
Loss of two RCP and > 55 percent power
Loss of one main feed pump and > 60 percent power
RC flow limit percent core power $\geq \frac{105(\text{RC flow})}{131.32 \times 10^6 \text{ lbs/hr}}$
Asymmetric rod 9" out of alignment with its group average rod position and > 60 percent power.

CAUSES:

RCP Trip
MFP Trip
Dropped Rod

AUTOMATIC ACTION:

Unit runs back to:
75 percent power on a loss of (1) RCP at a rate of 50 percent/min. ≈ 666 MWe
50 percent power on a loss of (2) RCP at a rate of 50 percent/min. ≈ 435 MWe
68 percent power on a loss of (1) MFP at a rate of 50 percent/min. ≈ 585 MWe
The RC flow limit at a rate of 20 percent min.
55 percent power on a Assy. rod fault at a rate of 30 percent/min. ≈ 482 MWe

NOTE
Runbacks are based on a MWe. Input power may vary due to plant efficiencies.

Form ES-401-6

Written Exam Question Worksheet

Q # 054

Examination Outline Cross-Reference

				RO	SRO
KA #	SYSTEM 061 K6.01	Page	Tier #	2	
RO/SRO Importance Rating	2.5	2.8	Group #	1	

Measurement Knowledge of the effect that a loss of the following will have on the AFW components: Controllers and positioners

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** D

The following sequence of events occurred:
- there was simultaneous loss of ICCW and the running Makeup pump
- the Reactor tripped automatically
- all three Emergency Feedwater pumps started
- EFW flow to "A" OTSG is 0 gpm
- EFW flow to "B" OTSG is 400 gpm

The following status indications are present above the EF-V-30 controllers for the "A" OTSG:
- AUTO light is illuminated
- SU light is illuminated
- OP light is not illuminated

Which ONE (1) of the following is the ONLY method which will allow the operator to control the correct level in the "A" OTSG?

A. Place EF-V-30A and/or D controllers in Local Auto and dial in a 25% setpoint.

B. Place EF-V-30A and/or D controllers in Local Auto and dial in a 50% setpoint.

C. Place EF-V-30A and/or D controllers in Manual and control level manually at 25".

D. Place EF-V-30A and/or D controllers in Manual and control level manually at 50%.

Technical Reference OPM Section F-10, Heat Sink Protection System, Rev. 9 pages 13 &16

Open Exam Reference

Learning Objective IV.E.05.04 Given a set of plant conditions and the status of the Emergency Feedwater system (EFW), determine if HSPS will actuate the EFW system, and what the response of the EFW system will be.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .7

Discriminant Validity Statements

A Plausible if the examinee does not diagnose the trip of all four reactor coolant pumps. range.

B Plausible if the examinee does not diagnose the failure to swap to the proper level

C Plausible if the examinee does not diagnose the trip of all four reactor coolant pumps.

D Correct answer.

Comments None.

FOR INFORMATION ONLY

SECTION F-10
REVISION 9

TMI OPERATIONS PLANT MANUAL

HEAT SINK PROTECTION SYSTEM

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Concur

Owner George J. Bifler Date 2/5/98

Operations Training Manager, TMI R. Pamfilo Date 2/16/98

Training Coordinator, TMI Linda L. Ritter Date 2-17-98

Approved

Plant Operations Director TMI L. G. Nol Date 2-25-98

d. Test Panel Components - Behind alarmed door panels are located 1) test switches, 2) test input jacks, 3) bypass switches, and 4) instrument selector switches. Test switches, bypass switches, and corresponding external indicators are listed on Table 1. Refer to Figure 13 to explain operation of test and bypass switches.

1. A test switch if operated places an individual channel function in the actuated state. To prevent switches from being left in this condition, switch covers ensure that the switch is properly placed prior to closing the test panel.

2. Analog input test jacks are provided where needed for calibrations

3. Bypass switches prevent channel actuation from inputting to the 2 out of 4 logic. The bypass switch does not bypass an actuated signal caused by a test switch.

4. Instrument selector switches located in the train cabinets are available to swap input instruments for EF-V-30 control (See Figure 7 & 8 and Table 1). An additional set of toggle switches is used to select which level signals will be used for console indication and potential ICS inputs.

e. Power Supplies - Power supplies were discussed in previous section.

f. External Indicators (See Table 1) - Indicating lights on the outside of Section A2 RK4 and Section A1 RK5 provide HSPS status locally for use during testing. These lights can also be used to confirm control room HSPS status indicators. There are (2) fuses for these lights located above the train test panel.

5.1.4 Logic and Actuation

The HSPS performs two types of actuation. It initiates EFW or isolates MFW.

EFW Initiation

There are two trains of EFW initiation. Both trains are designed to actuate following:

a. Lo OTSG Level (<10" S/U Level)

b. High RB pressure (>4 psig)

c. Lost of all 4 RCPS (pump power monitors)

d. Loss of both FWPS (FWP turb hyd oil pressure switches)

High RB pressure, loss of all 4 RCPS and loss of both FWPS will initiate EFW to both OTSG's. Low level actuation will only initiate EFW to the OTSG with low level. Train A when actuated starts EF-P-2A, opens MS-V-13A & B and transfer's EF-V-30A & C setpoints from 0" to 25". Train B starts EF-P-2B, opens MS-V-13A & B and transfer's EF-V-30B & D setpoint from 0" to 25". On loss of all RCPS, setpoint selection is to 50% operating range level instead of 25" startup range.

b. Control Room Lights and Indicators

Cont. Press		Hi	Hi	Hi	Hi	Lit if channel senses RB press >4 psig (see Figure 6)

OTSG A Level		Lo	Lo	Lo	Lo	Lit if channel senses OTSG level <10" (Pump Logic) (see Figure 1)
OTSG B Level		Lo	Lo	Lo	Lo	

OTSG A	Press	Lo	Lo	Lo	Lo	"Lo" light lit if channel senses press <600 (see Figure 5)
	Level	Hi Hi	Hi Hi	Hi Hi	Hi Hi	

OTSG B	Press	Lo	Lo	Lo	Lo	"Hi Hi" light lit if channel senses OTSG Level >97.5% (see Figure 4)
	Level	Hi Hi	Hi Hi	Hi Hi	Hi Hi	

Feed Pumps	Lit if both feed pump turbine switches sense low hyd. oil press. (<75 psig) (see Figure 3).
RC Pumps	Lit if all four RC pump power monitors sense low power (see Figure 2).

Auto		Lit if actuation signal is present or locked into related EF-V-30 controller.
SU	OP	"SU" vs "OP" indicates which type OTSG level is being displayed on the EF-V-30 controller.

5.2.2 Defeat/Enable (Refer to Figure 19)

These controls are normally operated in accordance with OP 1102-1, 1102-2, 1102-10 & 1102-11.

a. EFW Actuation Defeat/Enable Controls

There are Train A and B defeat/enable switches for each condition which initiates EFW. There are four associated with Train A on console left and four associated with Train B on console center. To completely defeat actuation due to any one of the following conditions both switches (Train A and B) must be placed in defeat.

❶ Loss of FW pumps
❷ Loss of all RCPs
❸ Lo Lo OTSG level (single DEF/EN switch for OTSG A or OTSG B)
❹ Hi RB pressure

Form ES-401-6

Written Exam Question Worksheet

Q # 055

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 061 GEN. 2.1.30 Page 2-4 Tier # 2

RO/SRO Importance Rating 3.9 3.4 Group # 1

Measurement Ability to locate and operate components, including local controls

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

The following plant conditions exist:
- Fifteen (15) minutes after a reactor trip from 100% power
- All four (4) Reactor Coolant Pumps are operating
- Emergency Feedwater was actuated on low OTSG levels
- "B" OTSG level is 55" and rising at 7" per minute
- EFW flow to "B" OTSG is 300 gpm and stable
- EF-V-30B and EF-V-30C controllers in the Control Room indicate 0 demand
- An AO reports that EF-V-30B is closed and EF-V-30C is 30% open

Which ONE (1) action below is required to gain control of "B" OTSG level?

A. Secure EF-P-2A from the Control Room.

B. Order the AO to manually close EF-V-52C.

C. Order the AO to manually close EF-V-30C with the handwheel.

D. Take manual control of EF-V-30C from the Control Room positioner.

Technical Reference ATOG 1210-10, Abnormal Transients Rules, Guides and Graphs, Rev. 37 page 13

Open Exam Reference

Learning Objective 061800050 1 Following an Emergency Feedwater Actuation, respond IAW ATP 1210-10, Abnormal Transients Rules, Guides and Graphs.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .7

Discriminant Validity Statements

A Plausible with misconception that EFW trains are split.

B Correct answer.

C Plausible because the valve does have a handwheel, however the handwheel can only open the valve.

D Plausible because this is an operator action, however the valve demand is already at 0%.

Comments None.

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TMI
Abnormal Transient Procedure

Number: 1210-10

Title: **Abnormal Transients Rules, Guides and Graphs**

Revision No.: 37

Applicability/Scope: TMI Division

Responsible Office: Plant Operations Dir.

Effective Date: 10/15/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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Originator	[illegible]	10-14-99
Procedure Owner	James R Faulks	10/15/99
PRG	[illegible]	10/15/99
pprover	James R Faulks	10/15/99

2.10.2 Failure of EF-P-2A or EF-P-2B to Start.

a. Verify that there is voltage available at the associated bus.

> NOTE
>
> With an ES signal present there will be a 20 second delay in pump start.

b. Verify that control power is available as indicated by the green indicator light at the control switch.

c. Verify all eight auto-defeat HSPS EFW initiation switches are in defeat. Manually start the pump using the control switch.

d. Check locally for targets on relays located at the switchgear and check the 10 amp closing fuses.

e. Use the 69 bypass key and attempt to start the pump locally.

2.10.3 Failure of EF-V-30A/D OR EF-V-30B/C to Open/Close.

a. Verify OTSG levels can be maintained.

b. Isolate failed EF-V-30 valve using the manual block valve EF-V-52A-D.

c. If the EF-V-30 valve is failed closed, THEN open one EF-V-30 to each OTSG to maintain required level.

d. If the EF-V-30 valve is failed open **AND** EF-V-52A-D cannot be operated due to a steam leak in the Intermediate Building **AND** isolation is required to mitigate an overcooling, **THEN:**

> NOTE
>
> To minimize dp across the EF-V-2, the EF-V-2 valve on the affected OTSG should be closed prior to securing the motor driven EFW pump. This isolation assumes an additional EFW pump is in operation.

1. Close the EF-V-2 on the affected OTSG, and

2. Secure the EF-P-2 to the affected OTSG.

2.10.4 No Indicated Flow (Low Flow) with Steam Generator Level below Setpoint.

a. Verify that EFP discharge pressure is higher than steam generator pressure. If not turbine header pressure setpoint may be reduced to get additional flow.

Form ES-401-6

Written Exam Question Worksheet

Q # 056

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 068 A2.04 Page 3.9-4 Tier # 2

RO/SRO Importance Rating 3.3 3.3 Group # 1

Measurement Ability to predict the impacts of the following malfunctions or operations on the Liquid Radwaste System; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Failure of automatic isolation

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

Plant Conditions:
- Reactor power is 100%.
- Liquid Release is in progress from A WECST.
- IWTS release is in progress.

A high alarm is received from RM-L-6 Radioactive Waste Water Discharge Monitor, however RM-L6 Enable/Defeat switch is in Defeat.

From the list below, identify the ONE (1) statement that describes required operator action(s) for this condition.

A. Close IW-V-73, IWTS Effluent valve on the Control Room LWDS panel.

B. Close WDL-V-257, Liquid Release Effluent Valve, on the Control Room LWDS panel.

C. Place RM-L6 to Enable AND verify IW-V-73, IWTS Effluent valve, closes.

D. Place RM-L6 to Enable AND verify WDL-V-257, Liquid Release Effluent Valve, closes.

Technical Reference EP 1202-12, Excessive Radiation Levels, Rev. 45, Pages 9 and 13.

Open Exam Reference

Learning Objective IV.B.09.04 Describe how to perform an authorized release of a W.E.C.S.T. to the effluent of the MDCT. Include all interlocks and automatic protection associated with discharges to the environment.

Question Source ☐ New ☑ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question # 76

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☑ 55.43 .5 ☑ 55.45 .3, .13

Discriminant Validity Statements

A Plausible since IW-V-73 closes on high radiation but from RM-L12 but is not operable from LWDS panel.

B Plausible since WDL-V-257 closes on high radiation from RM-L6 but is not operable from LWDS panel.

C Plausible since taking RM-L-6 to Enable would activate the interlock but does not close IW-V-7.

D Correct answer.

Comments None.

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TMI
Emergency Procedure

Number: 1202-12

Title: **Excessive Radiation Levels**

Revision No.: 45

Applicability/Scope: TMI Division

Responsible Office: Plant Operations Dir.

Effective Date: 10/11/99

This document is within QA plan scope [X] Yes [ ] No

Safety Reviews Required [X] Yes [ ] No

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Originator	[illegible]	10/2/99
Procedure Owner	[illegible]	10/2/99
PRG	[illegible]	10/5/99
Approver	James R. Paules	10/3/99

d. If radiation levels around the powdex vessels are >5 mr/hr at 10 ft due to an OTSG tube leak, notify GRCS, and back flush the vessels and re-coat the vessels with new resin.

e. If radiation levels around the powdex vessels or powdex recovery vessels are >100 mr/hr at 10 ft, shut down and cool down the unit to repair the OTSG tube leaks.

f. If Turbine Building general area exceeds 5 mr/hr, plant shutdown is required.

c. Liquid Monitor:

1. Alert Alarm:

a. Notify GRCS.

b. Obtain sample from affected system and analyze sample in accordance with approved Rad Con or Chemistry Procedures.

c. Take action as necessary to contain radioactive liquid.

d. Take appropriate action to reduce activity level.

e. If the alert alarm occurs on RM-L-7 or 12 complete Enclosure 2, "Operations Unplanned Release Report" if required.

2. Hi Alarm:

a. Notify GRCS.

b. Obtain sample from affected system and analyze sample in accordance with appropriate Rad Con or Chemistry Procedures. If analysis indicated activity confirming the alarm(s) for closed cooling water systems then isolate affected portion of system as necessary.

c. Obtain additional samples from same and/or additional locations and analyze count as required.

d. Take action as necessary to contain radioactive liquid.

e. Take appropriate action to reduce activity level.

ENCLOSURE 1

2.0 Interlock functions actuated by Hi Alarms from specific liquid radiation monitor are as follows:

_____ 2.1 RCS Letdown Line Monitor, RM-L-1 Hi.

RCS Letdown Isolation Valves MU-V-2A and MU-V-2B closes.

_____ 2.2 Radioactive Waste Water Discharge Monitor, RM-L6.

Mechanical Draft Cooling Tower Stop Valve, WDL-V257, closes.

_____ 2.3 Plant Discharge Line Monitor, RM-L7.

Mechanical Draft Cooling Tower Stop Valve, WDL-V257, closes.

_____ 2.4 IWTS/FS Discharge Line Monitor, RM-L12

The following pumps/valves trip off/shut.

a. IW-P-16, 17, 18, IW-V-73 - Effluent pumps/valve in IWTS.

b. IW-P-29, 30, IW-V-279 - Effluent pumps/valve in IWFS.

3.0 Interlock Functions actuated by Hi Alarms from specific area monitors

_____ 3.1 "A" O.T.S.G. Sample Line Monitor RM-G-16

"A" O.T.S.G. Sample Line Isolation Valves CA-V-4A and CA-V-5A close.

_____ 3.2 "B" O.T.S.G. Sample Line Monitor RM-G-17

"B" O.T.S.G. Sample Line Isolation Valves CA-V-4B and CA-V-5B close.

_____ 3.3 R.C.S. Sample Line Monitor RM-G-18

R.C.S. Sample Line Isolation Valves CA-V-1, CA-V-2, CA-V-3 and CA-V-13 close.

_____ 3.4 R.C. Drain Pump Discharge Line Monitor RM-G-20

a. R.C. Drain Pump Discharge Isolation Valves WDL-V-304 and WDL-V-303 close.

b. R.C. Drain Tank Vent Isolation Valves WDG-V-3 and WDG-V-4 close.

_____ 3.5 Reactor Building Sump Drain Monitor RM-G-21

Reactor Building Sump Drain Line Isolation Valves WDL-V-534 and WDL-V-535 close.

Form ES-401-6

Written Exam Question Worksheet

Q # 057

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 071 A3.03 Page 3.9-7 Tier # 2

RO/SRO Importance Rating 3.6 3.8 Group # 1

Measurement Ability to monitor automatic operation of the Waste Gas Disposal System including: Radiation monitoring system alarm and actuating signals

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

Waste Gas Decay Tank relief valve, WDG-V-36, has opened due to high tank pressure. This valve is now FAILED OPEN.

All the radiation monitors in the release pathway go into HIGH alarm.

From the list below, identify the ONE (1) statement that describes automatic action(s) initiated by the Radiation Monitoring system related to this accidental gaseous release.

A. Trips AH-E-14A/C (B/D).

B. Trips AH-E-10 AND AH-E-11.

C. Trips AH-E-10 ONLY.

D. Trips AH-E-11 ONLY.

Technical Reference EP 1202-12, Excessive Radiation Levels, Rev. 45, Page 12.
ARP MAP C-1-1, Radiation Level Hi, Rev. 72, Page 19.

Open Exam Reference

Learning Objective IV.E.06.02 Identify the radiation monitoring channels by letter/number designation and the system/area monitored by each channel.

Question Source ☐ New ☑ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question # 32

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .5

Discriminant Validity Statements

A Plausible misconception that reduction of ventilation exhaust flow will terminate or reduce gaseous radwaste releases.

B Correct answer.

C Plausible distracter since this is the supply fan for the Fuel Handling Building.

D Plausible distracter since this is the supply fan for the Auxiliary Building, where this accidental release originates.

Comments None.

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GPU NUCLEAR

TMI
Emergency Procedure

Number: 1202-12

Title: **Excessive Radiation Levels**

Revision No.: 45

Applicability/Scope: TMI Division

Responsible Office: Plant Operations Dir.

Effective Date: 10/11/99

This document is within QA plan scope: [X] Yes [ ] No

Safety Reviews Required: [X] Yes [ ] No

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Originator	[illegible]	10/2/99
Procedure Owner	[illegible]	10/2/99
PRG	[illegible]	10/5/99
Approver	James R. Paulas	10/3/99

_____ 1.3 Condenser Vacuum Pump Discharge Monitor RM-A5 Low Range/15 High Alarm initiates the MAP-5 Iodine-Particulate sequential sampling.

_____ 1.4 Auxiliary Building Ventilation Duct Monitor (gas channel) RM-A6:

Auxiliary Building Supply Fan (AH-E-11) shuts down.

_____ 1.5 Gas Waste Tank Discharge Monitor, RM-A7:

Waste Gas Release Valve, WDG-V47, closes.

_____ 1.6 Auxiliary and Fuel Handling Building Exhaust Duct (gas channel), RM-A8.

a. MAP-5 Iodine-Particulate sampling is initiated upon high alarm on RM-A-8 gas channel.

b. The following fans shutdown:

1. Fuel Handling Building Supply Fan (AH-E-10).

2. Auxiliary Building Supply Fan (AH-E-11).

c. Waste Gas Release Valve, WDG-V47, closes.

_____ 1.7 Reactor Building Purge Duct (gas channel), RM-A9:

a. MAP-5 Iodine-Particulate sampling is initiated upon RM-A-9 high alarm on gas channel.

b. The following valves close:

1. Outside Reactor Building Purge Exhaust, AH-V1A.

2. Inside Reactor Building Purge Exhaust, AH-V1B.

3. Inside Reactor Building Purge Supply, AH-V1C.

4. Outside Reactor Building Purge Supply, AH-V1D.

5. Reactor Building Sump Outlet Isolation, WDL-V534.

6. Reactor Building Sump Outlet Isolation Valve, WDL-V535.

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GPU NUCLEAR	TMI Alarm Response Procedure	Number: MAP C
Title: Main Annunciator Panel C		Revision No.: 72
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: 09/07/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

TRAINING DEPT.

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	Signature	Date
Originator	R E Bay	8-26-99
Procedure Owner	R E Bay	8-26-99
PRG	[illegible]	8/31/99
Approver	[illegible]	8/27/99

ALARM:

RM-A-8 AUX. AND F.H.B. EXH. DUCT

SET POINTS:

Refer to Operating Procedure 1101-2.1, RMS Setpoints

CAUSES:

1. Airborne radioactivity in the Aux. Bldg. and/or F.H.B.
2. Hi gas release activity from W.G. Decay Tks.

AUTOMATIC ACTION:

1. Gas release will be terminated by closing WDG-V-47.
2. AH-E-10 and 11 will trip on a gaseous Hi Alarm.

OBSERVATION (CONTROL ROOM):

1. RM-A-8 on PRF Alert Alarm (yellow light)
2. RM-A-8 on PRF Hi Alarm (red light)
3. RM-A-8 Indicator > one or more of the above set points

MANUAL ACTION REQUIRED:

Alert:

1. Determine which of RM-A-4, 6 and/or 7 are alarming, follow manual action on appropriate alarm response.
2. Notify GRCS.

Hi Alarm:

1. Verify AH-E-10, 11 trip and WDG-V-47 closed.
2. Have Chemistry take RM-A8 samples to determine source term.
3. Determine which of RM-A-4, 6 and/or 7 are alarming and follow manual action on appropriate alarm response.
4. Refer to Emergency Procedure 1202-12 Excessive Rad Levels.

Form ES-401-6

Written Exam Question Worksheet

Q # 058

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 072 K5.02 Page 3.7-13 Tier # 2

RO/SRO Importance Rating 2.5 3.2 Group # 1

Measurement Knowledge of the operational implications of the following concepts as they apply to the ARM system: Radiation intensity changes with source distance

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

Which ONE (1) of the following explains the relationship between the dose rate from a POINT radiation source and the distance from that source?

A. Dose rate is proportional to the change in distance.

B. Dose rate is inversely proportional to the change in distance.

C. Dose rate is proportional to the change in distance squared.

D. Dose rate is inversely proportional to the change in distance squared.

Technical Reference OPM Section O-03, Radiation Control and Safety, Rev.6 pages 58-60

Open Exam Reference

Learning Objective III.F.01.09 Explain how time, distance and shielding can be used to minimize radiation exposure.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☐ 55.43 ☑ 55.45 .7

Discriminant Validity Statements

A Plausible for a plane source but misses the inverse relationship.

B Plausible for a plane source.

C Plausible, but misses the inverse relationship.

D Correct answer.

Comments None.

FOR INFORMATION ONLY

TMI OPERATIONS PLANT MANUAL

RADIATION CONTROL AND SAFETY

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21	6	46	6	71	6		
22	6	47	6	72	6		
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25	6	50	6	75	6		

Concur

Owner — Dennis L. Fox — Date 9-8-98

Operations Training Manager, TMI — N. L. Orth Jr. — Date 9/10/98

Training Coordinator, TMI — Linda L. Ritter — Date 9-15-98

Approved

Plant Operations Director TMI — James Naulls for Nau — Date 9/15/98

❷ Calculation of Time and Doses Involving More Than One Dose Rate

These calculations combine the calculation done in (❶) above with the type done in Section E-3.

Example: You have been working at a dose rate of 50 mrem/hr for 20 minutes. You are beginning to work in a new location where the dose rate is 200 mrem/hr. How much time do you have before you receive a total dose of 50 mrem?

First, calculate the dose received so far:

20 min = 1/3 or 0.33 hrs

Dose = Dose Rate x Time
= 50 mrem/hr x 0.33 hr
= 16.7 or 17 mrem.

Second, calculate how much remains of the dose you are to receive that day:

50 - 17 = 33 mrem which can still be received.

Third, calculate the time required at the new dose rate to receive this remaining dose:

$$\text{Time} = \frac{\text{Dose}}{\text{Dose Rate}}$$

$$= \frac{33 \text{ mrem}}{200 \text{ mrem / hr}}$$

$$= 0.165 \text{ hrs}$$

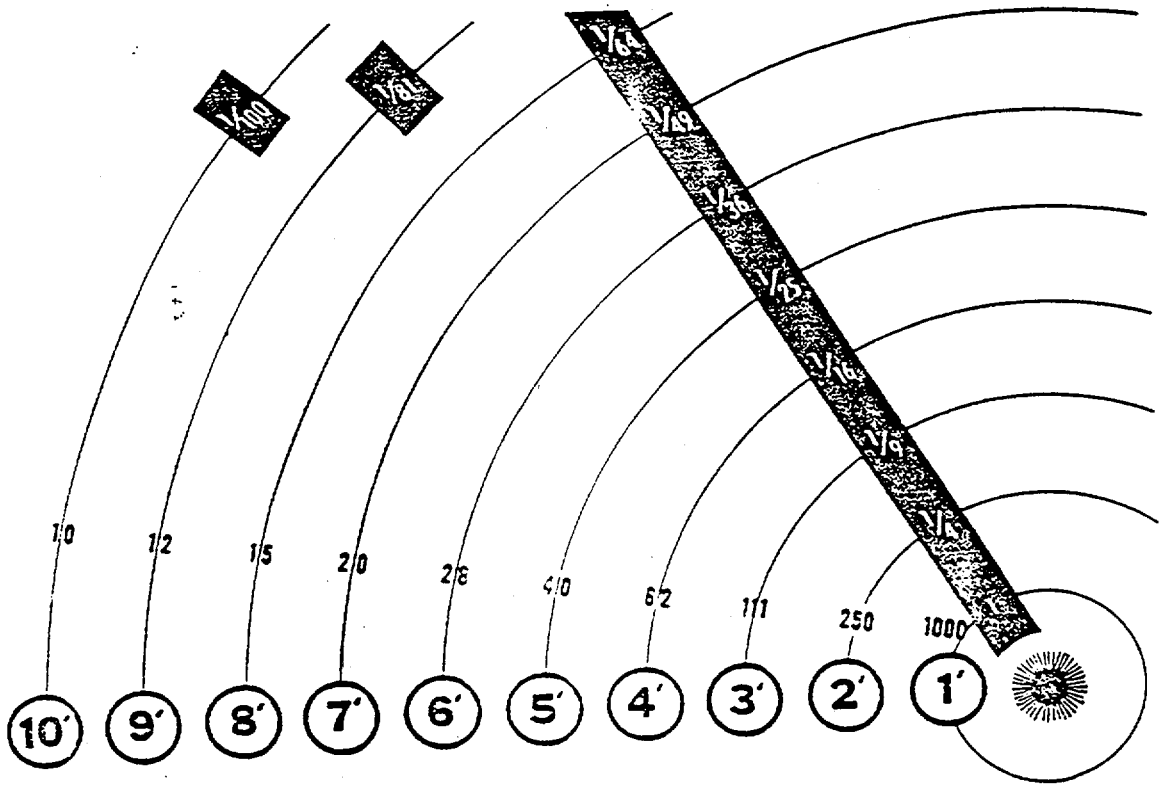
$$= 0.165 \text{ hrs} \times \frac{60 \text{ min}}{\text{hr}}$$

$$= 9.9 \text{ min}$$

b. Distance

The effect of distance on radiation exposure is quite startling if the source is a point source. This effect is due to the inverse square law, i.e., the intensity of radiation falls off by the square of the distance from the source, as shown in Figure 7.

FIGURE 7 - EFFECT OF DISTANCE ON RADIATION EXPOSURE



It should be emphasized that the inverse square law applies to the degree that distances are large in relation to the size of the source. Radiation sources used in industry for penetrating radiation are generally quite small in size, so the inverse square law can be applied to distance in the immediate vicinity of the source. If, however, the source is large in size, such as the side of a water storage tank, which covers a considerable area, or as might be the case if the radioactive material were dispersed, then the inverse square law does not start to apply until the distances are large in relation to the size of the source. For the inverse square law to apply, the source must be a gamma or neutron source, and the distance from the source must be at least three times the length of the largest dimension of the source. If the distance from the source does not meet this point source criteria, then the dose rate will not change as rapidly with distance.

If we had a point source of gamma radiation giving us 1,000 units of penetrating external radiation at 1 foot, we would receive only 250 units at 2 feet because we have doubled the distance, and the effect on the radiation level is to reduce it to $(1/2)^2$ or 1/4. When we have tripled the distance, we have reduced the dose to $(1/3)^2$ or 1/9th, to 111 units and so on. At 10 feet away, we have $(1/10)^2$ or 1/100th of the radiation exposure at 1 foot, i.e., 10 units.

Look at the distance chart now in terms of actual doses from a point source and their possible effect on a man. Consider that the radiation rate at 1 foot is 1,000 R per hour, and assume that a man remains at that distance for 1 hour from our point source. He will receive 1,000 rem, an absolutely lethal dose of radiation. If he remains 2 feet from the source for 1 hour, he will receive 250 rem. This puts him on the threshold of dying, and he might be extremely ill. At 3 feet, he would receive only 111 rem, and we would not expect that he would die, but he might be sick for a period of time.

At 4 feet, he is already in a range where there would probably be no symptoms that he would detect, and the only damage that we could find would be certain changes in his blood, which would reverse themselves with the passage of time. If we extended our chart out to 20 feet, we would see that the man at 20 feet would get only 2 1/2 rem; this is 1/400th $(1/20)^2$ of the dose at 1 foot, less than the amount permitted to a radiation worker in a 3-month period.

We see, therefore, that there is a dramatic fall-off in the rate of radiation exposure as we go away from a point source of the radiation, and a very little bit of distance can go a long way in increasing our safety factor. Conversely, as we close in on the radiation source, our levels will go higher by the same mathematical process. For instance, if we move in to 6 inches, the radiation rate will go from 1,000 R per hour to 4,000 R per hour. This helps to make us realize one of the most fundamental rules of radiation protection -- to maintain the maximum possible distance between ourselves and the source of the radiation. We should avoid touching materials which give off significant amounts of penetrating external radiation, because, in such a case, we have obviously no distance at all working for us.

The inverse square law can be written as follows:

I_o (Dose Rate at Distance d_o) x d (Original distance)^2 =

I_f (Dose Rate at Distance d_f) x d (Final Distance)^2

so that if you know any three items in this equation you can find the fourth.

$$I_o \times d = I_f \times d$$

If the use of this formula is confusing to you, the following method can be used:

- Take the two distances involved (making sure each is measured from the location of the source) and divide the larger by the smaller.
- Square the result to obtain the "distance factor."
- Using logical reasoning, either multiply or divide the given dose rate by the distance factor. (If you are going closer to the source, the radiation level will go up, so you multiply; if you are moving farther away from the source, the radiation level should decrease, so you divide by the factor.)

Example: The dose rate from a small valve is 30 mR/hr at 10'. What would you expect the dose rate to be 2' from the valve?

- $\frac{\text{Larger Distance}}{\text{Smaller Distance}} = \frac{10}{2} = 5$
- Square it = 25
- Since we are going closer, the radiation level will increase, so we multiply by the factor: 30 mR/hr x 25 = 750 mR/hr at 2'

Form ES-401-6

Written Exam Question Worksheet

Q # 059

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 072 GEN. 2.4.2 Page 2-11 Tier # 2

RO/SRO Importance Rating 3.9 4.1 Group # 1

Measurement Knowledge of system setpoints, interlocks and automatic actions associated with EOP entry conditions

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

Which ONE (1) of the following will occur when RM-A-9G goes into HI alarm?

A. WDG-V47 closes.

B. WDL-V535 closes.

C. AH-E-10 trips.

D. AH-E-11 trips.

Technical Reference EP 1202-12, Excessive Radiation Levels, page 12, Rev. 45.

Open Exam Reference

Learning Objective V.D.07.02 State the Immediate Automatic Actions that occur following an Excessive Radiation Levels event given the monitor that went into alarm per EP 1202-12.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .7, .8

Discriminant Validity Statements

A Plausible since this action occurs for other rad monitors.

B Correct answer.

C Plausible since this action occurs for other rad monitors.

D Plausible since this action occurs for other rad monitors.

Comments None.

FOR INFORMATION ONLY

GPU NUCLEAR	TMI Emergency Procedure	Number Q-059 1202-12
Title: **Excessive Radiation Levels**		Revision No. 45
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: 10/11/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	Allen Yontz	10/2/99
Procedure Owner	[illegible]	10/2/99
PRG	[illegible]	10/5/99
Approver	James R Paulas	10/3/99

_____ 1.3 Condenser Vacuum Pump Discharge Monitor RM-A5 Low Range/15 High Alarm initiates the MAP-5 Iodine-Particulate sequential sampling.

_____ 1.4 Auxiliary Building Ventilation Duct Monitor (gas channel) RM-A6:

Auxiliary Building Supply Fan (AH-E-11) shuts down.

_____ 1.5 Gas Waste Tank Discharge Monitor, RM-A7:

Waste Gas Release Valve, WDG-V47, closes.

_____ 1.6 Auxiliary and Fuel Handling Building Exhaust Duct (gas channel), RM-A8.

a. MAP-5 Iodine-Particulate sampling is initiated upon high alarm on RM-A-8 gas channel.

b. The following fans shutdown:

1. Fuel Handling Building Supply Fan (AH-E-10).

2. Auxiliary Building Supply Fan (AH-E-11).

c. Waste Gas Release Valve, WDG-V47, closes.

_____ 1.7 Reactor Building Purge Duct (gas channel), RM-A9:

a. MAP-5 Iodine-Particulate sampling is initiated upon RM-A-9 high alarm on gas channel.

b. The following valves close:

1. Outside Reactor Building Purge Exhaust, AH-V1A.

2. Inside Reactor Building Purge Exhaust, AH-V1B.

3. Inside Reactor Building Purge Supply, AH-V1C.

4. Outside Reactor Building Purge Supply, AH-V1D.

5. Reactor Building Sump Outlet Isolation, WDL-V534.

6. Reactor Building Sump Outlet Isolation Valve, WDL-V535.

Form ES-401-6

Written Exam Question Worksheet

Q # 060

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 002 GEN.2.2.25 Page 2-7 Tier # 2

RO/SRO Importance Rating 2.5 3.7 Group # 2

Measurement Knowledge of bases in Tech Specs for LCOs and safety limits

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

Which ONE (1) of the following is the basis the Reactor Protection System Nuclear Overpower trip setpoint in Technical Specifications?

A. To prevent excessive core outlet temperature in the operating range.

B. To protect against a startup accident from low power or a slow rod withdrawal from high power.

C. To accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power.

D. To prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

Technical Reference Tech. Specs. Page 2-5

Open Exam Reference

Learning Objective IV.E.14.06 List the automatic reactor trips including setpoints and state the bases for each as found in Tech Specs.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .2 ☐ 55.45

Discriminant Validity Statements

A Plausible since core outlet temperature increases with power at higher power levels. This is the basis for the core outlet temperature trip.

B Plausible due to mention of slow rod withdrawal from high power. This is the basis for high reactor coolant system pressure.

C Plausible due to mention of high power. This is the basis for the overpower trip based on flow and imbalance.

D Correct answer.

Comments None.

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2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTION INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, axial power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protection action to prevent any combination of process variables from exceeding a safety limit.

Specification

2.3.1 The reactor protection system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1 and the Protection System Maximum Allowable Setpoints for Axial Power Imbalance as given in the COLR.

Bases

The reactor protection system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protection system instrumentation are listed in Table 2.3-1. These trip setpoints are setting limits on the setpoint side of the protection system bistable comparators. The safety analysis has been based upon these protection system instrumentation trip set points plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operations with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.1% of rated power. Adding to this the possible variation in trip set points due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis (Reference 1).

Amendment No. 13, 17, 28, 126, 142, 157, 184,

Form ES-401-6

Written Exam Question Worksheet

Q # 061

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 006 K3.03 Page 3.2-17 Tier # 2

RO/SRO Importance Rating 4.2 4.4 Group # 2

Measurement Knowledge of the effect that a loss or malfunction of the ECCS will have on the following: Containment

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** B

The plant was at 100% power in normal alignment when the following sequence of events occurred:
- small break LOCA
- RB pressure rose to 10 psig and stabilized
- 1600#, 500# RCS pressure ESAS signals were bypassed
- 4# RB pressure ESAS signal was defeated
- the LOCA degraded to a large break
- RB pressure rose to 40 psig

Which ONE (1) of the following identifies the equipment operating in order to limit containment pressure and temperature?

A. All three AH-E-1s in FAST speed.

B. All three AH-E-1s in SLOW speed.

C. All three AH-E-1s in FAST speed, AND both BS-P-1s.

D. All three AH-E-1s in SLOW speed, AND both BS-P-1s.

Technical Reference OPM Section B-09, Reactor Building Spray, Rev. 4 page 9

Open Exam Reference

Learning Objective IV.E.24.07 State the conditions that will auto start a Building Spray Pump.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .6

Discriminant Validity Statements

A Plausible because higher fan speed would provide more cooling.

B Correct answer.

C Plausible because of normal response of BS-P-1s and higher fan flow.

D Plausible because this would this would be the normal response.

Comments None.

FOR INFORMATION ONLY

SECTION B-09
REVISION 4

TMI OPERATIONS PLANT MANUAL
REACTOR BUILDING SPRAY

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Concur

Owner _____ Date 4-10-97

Operations Training Manager, TMI _____ Date 4/14/97

Training Coordinator, TMI _____ Date 4-15-97

Approved

Plant Operations Director TMI L. G. Nole Date 4-15-97

8.0 INTERFACING SYSTEMS

8.1 Decay Heat Removal

The Reactor Building Spray Pumps suction taps off the Decay Heat Removal System on the suction lines of the Decay Heat Removal Pumps.

8.2 Engineered Safeguards Actuation System

The 4# ESAS signal opens the Building Spray Valves and gives a start permissive signal to the Building Spray Pumps. The 30# ESAS signal gives an actuation signal to the Building Spray Pumps to start.

8.3 Heat Trace

The NaOH Tank is heat traced so that temperature in the tank is maintained between 50°F and 120°F. The piping associated with the NaOH Tank is also heat traced to insure the liquid in the piping does not solidify.

9.0 TECHNICAL SPECIFICATIONS

9.1 The following components in the Reactor Building Spray System must be operable.

9.1.1 Two Reactor Building Spray Pumps and their associated spray nozzles headers.

9.1.2 The Sodium Hydroxide Tank level shall be maintained at 8 feet $\pm$ 6 inches lower than the BWST level as measured by the BWST/NaOH Tank differential pressure indicators. The installed differential pressure indicator is located in the BWST tunnel at the 302' elevation and it should read 30 $\pm$ 5 standard inches of water. The NaOH Tank concentration shall be 10.0 $\pm$ .5 weight percent.

9.1.3 All manual valves in the discharge lines of the Sodium Hydroxide Tank shall be locked open.

9.2 Maintenance shall be allowed during power operation on any component in the Reactor Building Spray System which will not remove more than one train of the system from service. Components shall not be removed from service such that the entire train is inoperable for more than 72 hours. If the train removed from service is not restored to service within 72 hours, the reactor shall be placed in hot shutdown within six hours.

9.3 Prior to initiating maintenance or testing on any component:

9.3.1 The SF/SS must be notified prior to defeating and testing any portion of the Safety System.

9.3.2 The redundant component shall be verified to be operable (Tech. Spec. 3.3.4).

Form ES-401-6

Written Exam Question Worksheet

Q # 062

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 010 A2.02 Page 3.3-8 Tier # 2

RO/SRO Importance Rating 3.9 3.9 Group # 2

Measurement Ability to predict the impacts of the following malfunctions or operations on the PZR PCS system; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Spray valve failures

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

The following plant conditions currently exist:
- RCS pressure is 2125 psig and decreasing at 21 psig per minute
- Pressurizer temperature is 644°F and decreasing at 2.5°F per minute
- The Pressurizer Spray Valve (RC-V-1) has NO indication
- The PORV (RC-RV-2) has a green demand indication
- There is no evidence of RCS leakage.
- No operator actions have been taken.

Which ONE (1) of the following automatic OR manual actions is REQUIRED to prevent a low RCS pressure reactor trip?

A. Pressurizer Heater Bank 4 energizing at 2120 psig will mitigate the event.

B. Pressurizer Heater Bank 5 energizing at 2105 psig will mitigate the event.

C. The operator MUST close the Pressurizer Spray Block Valve (RC-V-3).

D. The operator MUST close the PORV Block Valve (RC-V-2).

Technical Reference EP 1202-29, Pressurizer System Failure, Rev.54 page 7

Open Exam Reference Steam Tables

Learning Objective V.D.11.03 Given a list of symptoms, identify the event, determine what Immediate Automatic Actions will occur and what Immediate Manual Actions are required IAW EP 1202-29.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☑ 55.43 .5 ☑ 55.45 .3, .13

Discriminant Validity Statements

A Plausible because this is an automatic action and the examinee may predict that this will overcome the effects of pressurizer spray.

B Plausible because this is an automatic action and the examinee may predict that this will overcome the effects of pressurizer spray.

C Correct answer.

D Plausible because this is the manual action required for a PORV failure.

Comments None.

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GPU NUCLEAR

TMI
Emergency Procedure

Number: 1202-29

Revision No.: 54

Pressurizer System Failure

Applicability/Scope	Responsible Office	Effective Date
TMI Division	Plant Operations Dir.	07/06/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

SIMULATOR ROOM
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	Signature	Date
Originator		6-21-99
Procedure Owner		6-24-99
PRG		6/29/99
Approver	James R Faulks	6/25/99

SECTION C

Decreasing or Low RCS Pressure

C.1 Symptoms

1. Decreasing RCS pressure on Control Room indicators, recorders and computer.
2. Decreasing PZR level.
3. Decreasing RCS temperature.
4. Decreasing PZR temperature.
5. Open indications for PORV, Code Safeties (Tailpipe ΔT A517, A518, A519/Elbow Tap ΔP/Acoustics Monitor Ann G-1-7, PORV Open [Acoustic])
6. PZR heaters fail to energize at desired setpoint.

	BK1	BK2	BK3	BK4	BK5
FULL ON	2135	2135	2135	2120	2105
OFF	2155	2155	2147	2140	2125

C.2 Immediate Action

A. Automatic Action

1. RC-V-1 closed at ≤ 2155 psig.
2. All PZR heaters are "ON" at < 2105.
3. RPS trips reactor at < 1900 psig.
4. ESAS actuation at < 1600 psig.

B. Manual Action

1. Verify PORV and Code Safeties are closed.
2. Block PORV (close RC-V-2) if it failed open.
3. Verify PZR spray valve (RC-V-1) is shut if press < 2155 psig.
4. Close RC-V-3 if RC-V-1 is failed open.
5. Verify pressurizer heaters energize at setpoint or manually turn "ON" additional heaters as necessary to maintain RCS pressure at ≥ 2135.
6. Monitor pressurizer level and RCS temp.

Form ES-401-6

Written Exam Question Worksheet

Q # 063

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 011 K1.02 Page 3.2-21 Tier # 2

RO/SRO Importance Rating 3.7 3.8 Group # 2

Measurement Knowledge of the physical connections and/or cause-effect relationships between the PZR LCS and the following systems: RCS

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

Which ONE (1) of the following describes the purpose of the minimum bypass flow around normal makeup valve MU-V-17?

A. Prevent thermal shock of the "B" HPI nozzle.

B. Prevent thermal shock of the Pressurizer surge line.

C. To ensure Pressurizer and RCS Boron concentrations remain equalized.

D. To ensure Makeup Tank and RCS Boron concentrations remain equalized.

Technical Reference OPM Section B-05, Makeup and Purification, Rev. 16 page 16

Open Exam Reference

Learning Objective IV.A.09.05 Briefly explain the purpose of minimum (bypass) makeup flow around MU-V-17 and describe where and how this flow is established.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .2 TO .9 ☐ 55.43 ☑ 55.45 .7, .8

Discriminant Validity Statements

A Correct answer.

B Plausible with the misconception that this minimum bypass flow will cause RCS flow into the pressurizer.

C Plausible, but this is one of the purposes of the pressurizer spray bypass flow.

D Plausible because this would be a side-effect.

Comments None.

FOR INFORMATION ONLY

SECTION B-05
REVISION 16

TMI OPERATIONS PLANT MANUAL
MAKEUP AND PURIFICATION

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Concur

Owner _____ Date 11/10/99

Operations Training Manager, TMI _____ Date 11/12/99

Training Coordinator, TMI _____ Date 11-16-99

Approved

Plant Operations Director TMI _____ Date 11/17/99

❹ To provide high pressure injection flow in case of a leak (Automatic) (See Emergency Operation VIII E.2). In situation (a) above the pump start will normally provide a transient in makeup flow with a subsequent return to its initial value. In situation (b), makeup flow will then increase as the second pump is started. In situation (c), where Pressurizer level is low, the makeup flow increase will be permanent. As in all pump starts in <u>any</u> system the operator should observe flow indication to ensure that the system response is as expected. Failure of the system to respond as expected is indicative of problems with the pump or an incorrect or abnormal valve lineup.

Normal makeup to the Primary Coolant loop is through MU-V-17, or Manual Throttle Valve MU-V-92, which controls flow to Loop "A" and maintains Pressurizer level.

A small line is connected from upstream of MU-V-17 to the thermal sleeve in the "B" Reactor Coolant Pump discharge piping. MU-V-205 is set for 3 gpm flow to provide a continuous flow through the line even when MU-V-17 is closed. This is to prevent thermal shock.

k. Makeup Pump/Speed Changer Oil Pumps

❶ Pump End - Two lube oil pumps supply oil to the motor air cooler and pump bearings for lubrication.

Operation: When Makeup Pump A or C extension control switch in the Control Room is placed in PULL-TO-LOCK (PTL) the Lube Oil Pumps are locked out. When taken out of PULL-TO-LOCK and placed in NORMAL-AFTER-START, both Main and Auxiliary Oil Pumps start and the LOW LUBE OIL PRESSURE alarm clears. The Auxiliary Oil Pump stops at 15 psig oil pressure increases. The main pump continues to run until the Makeup Pump is placed in PTL or started. MU-P-1B Oil Pumps are not controlled from MU-P-1B control switch. The main runs if its breaker is closed. The auto runs if its breaker is closed and the main pump discharge is <15 psi. The oil is cooled by NSCCW in all 3 pumps or by DHCCW in MU-P-1A/C. Short term cooling (<4 Hrs) of MU-P-1A/C is supplied from DHCCW. If either Makeup Pump will be operated >4 Hrs its cooling water supply should be shifted over to NSCCW. Due to the NSCCW system being pressurized, we do not cross connect NSCCW with DHCCW. For a short period of time there is no cooling flow through the heat exchanger. The discretion of the Shift Supervisor/Shift Foreman is required in shifting cooling water supplies while the pump is running. Refer to OP 1104-2 for actual procedure.

The Heat Exchanger is located in the Makeup Pump room near the motor end of the Makeup Pump.

❷ Speed Changer - Two Lube Oil Pumps supply oil to the speed changer bearing. One pump is AC powered and the other is a gear driven oil pump attached to the shaft.

Form ES-401-6

Written Exam Question Worksheet

Q # 064

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 012 A1.01 Page 3.7-3 Tier # 2

RO/SRO Importance Rating 2.9 3.4 Group # 2

Measurement Ability to predict/monitor changes in parameters associated with operating the RPS controls including: Trip setpoint adjustment

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

Which ONE (1) of the following sets of trips are ALL bypassed by the RPS cabinet Shutdown Bypass Key Switch?

A. Power to Pumps
Power/Imbalance/Flow
Variable Pressure/Temperature

B. Power/Imbalance/Flow
Reactor Building Pressure
Low Reactor Coolant Pressure

C. Loss of 2 Main Feed Pumps
Low Reactor Coolant Pressure
Variable Pressure/Temperature

D. Power to Pumps
Low Reactor Coolant Pressure
High Reactor Coolant Pressure

Technical Reference OPM Section F-02, Reactor Protection System, Rev. 9 page 119

Open Exam Reference

Learning Objective IV.E.14.11 Given plant conditions and the current operating status of the RPS (Shutdown Bypass or Normal mode), determine whether the four RPS channels should be tripped or reset.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☐ 55.43 ☑ 55.45 .5

Discriminant Validity Statements

A Correct answer.

B Plausible since all trips are in the RPS trip string.

C Plausible since all trips are in the RPS trip string.

D Plausible since all trips are in the RPS trip string.

Comments None.

FOR INFORMATION ONLY

SECTION F-02
REVISION 9

TMI OPERATIONS PLANT MANUAL
REACTOR PROTECTION SYSTEM

Q-64

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25	8	50	8	75	8	100	8

Concur

Owner Stuart Brontley Date 11-19-99

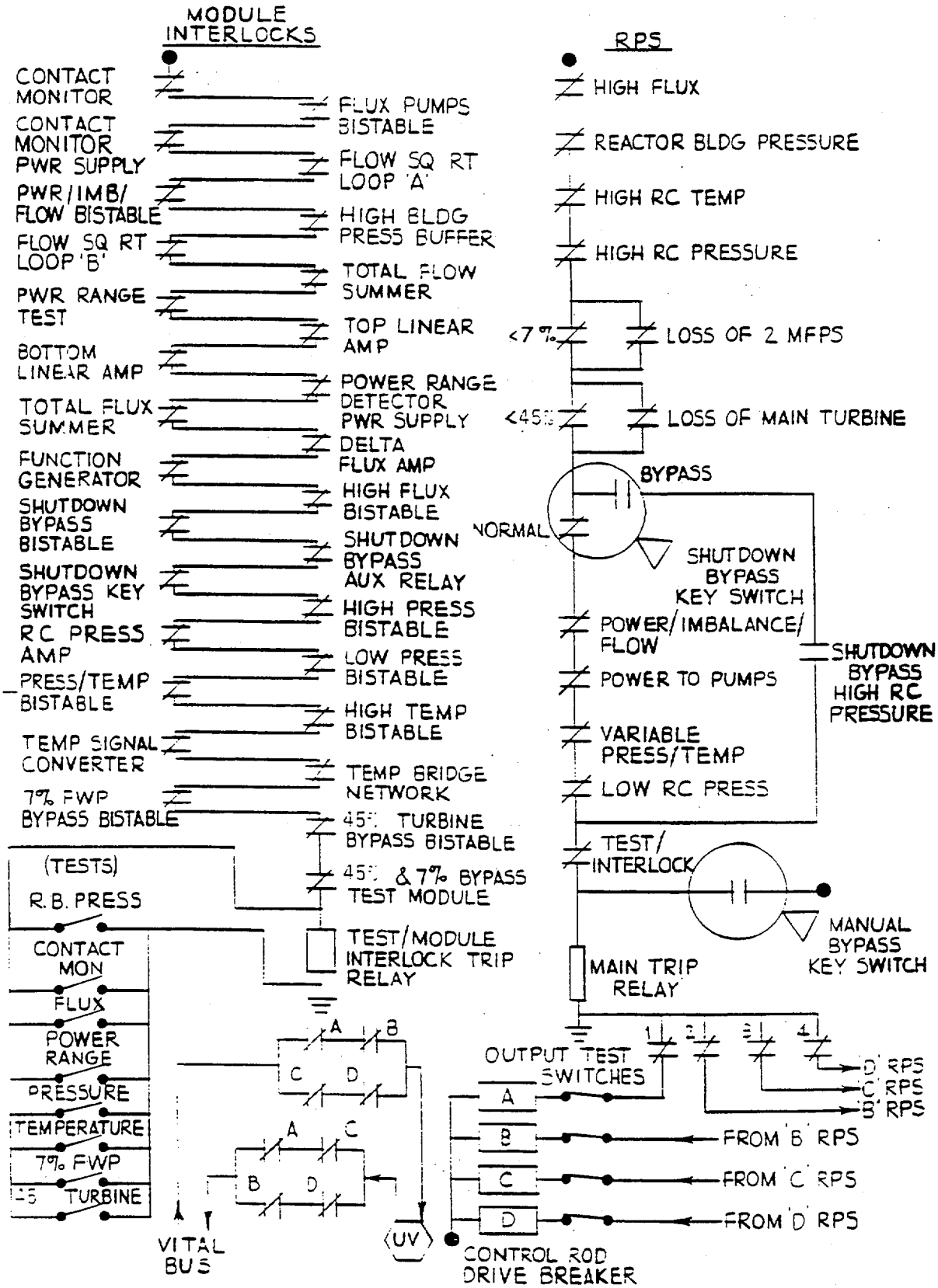
Operations Training Manager, TMI R L Lamb Date 12/1/99

Training Coordinator, TMI Linda L. Butler Date 12-2-99

Approved

Plant Operations Director TMI D W [illegible] Date 12/3/99

FIGURE 62 - INTERLOCK STRING AND TRIP CIRCUITS



Form ES-401-6

Written Exam Question Worksheet

Q # 065

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 014 K4.05 Page 3.1-21 Tier # 2

RO/SRO Importance Rating 3.1 3.3 Group # 2

Measurement Knowledge of RPIS design features and/or interlocks provide for the following: Rod hold interlocks

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

During an approach to criticality, the following plant conditions exist:
- NI-11 and 12 indicate 150 cps
- NI-3 and 4 indicate 1x10-11 amps

Which ONE (1) of the following would occur if NI-4 Startup Rate fails high while the operator is withdrawing rods?

A. Rod out motion would stop immediately.

B. An out-inhibit would go into effect when NI-4 comes on scale.

C. An out-inhibit would go into effect when the operator stops withdrawing rods.

D. Rod out motion would not be affected unless either NI-11 or NI-12 indicated a high startup rate.

Technical Reference OPM Section F-01, Control Rod Drive System, Rev. 7 page 62

Open Exam Reference

Learning Objective IV.E.11.07 For each Nuclear Instrument channel, identify its control functions and interlocks.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☐ 55.43 ☑ 55.45 .7

Discriminant Validity Statements

A Correct answer.

B Plausible with the misconception that the instrument must be within it's operating range.

C Plausible with the misconception that the current rod motion would be allowed.

D Plausible with the misconception of the logic circuit for the high SUR out-inhibit.

Comments None.

FOR INFORMATION ONLY

SECTION F-01
REVISION 7

TMI OPERATIONS PLANT MANUAL

CONTROL ROD DRIVE SYSTEM (ELECTRICAL AND MECHANICAL)

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24	7	49	7	74	7		
25	7	50	7	75	7		

Concur

Owner _____ Date 12-4-98

Operations Training Manager, TMI _____ Date 12/9/98

Training Coordinator, TMI _____ Date 12-11-98

Approved

Plant Operations Director TMI _____ Date 12-15-98

❺ Group 7 IN LIMIT with Group 6 more than 80% withdrawn and a 9 inch fault present.

b. As discussed earlier, an IN LIMIT for a group will stop that group's IN motion. This function must be overridden in the event of a dropped rod so that the group with the faulted rod can run back. This is accomplished by the fault bypass circuit.

❶ If the system is in AUTO and a 9 inch fault occurs, the fault bypass relay energizes, closing a contact that bypasses the group IN LIMIT contacts in the motor control circuitry.

❷ With the safety rods not withdrawn, it may be necessary to bypass that feature allow rod withdrawal. This is accomplished by a KEYLOCK BYPASS switch that bypasses the SAFETY RODS OUT circuit.

c. The Asymmetric Fault for a rod may be bypassed by using a switch located in that rod's PI Cabinet. This switch eliminates that rod from being used in the Asymmetric Fault Detection circuitry.

12.3 MOTOR FAULT Lamp

The MOTOR FAULT lamp indicates that one or more of the programmers has:

a. Any motion with no command,
b. OUT motion with an IN command,
c. Loss of 120V AC power.

❶ A MOTOR FAULT will reset the Rod Control System to manual and will lock in; once the malfunction has been corrected, the motor fault can be cleared with the fault reset switch on the Diamond Panel.

❷ The Motor Fault circuit is an electro-mechanical device. The direction is determined by magnetic switches, which are actuated by magnets driven directly by the programmer.

12.4 OUT INHIBIT Lamp

The OUT INHIBIT lamp indicates that the control rods will not respond to OUT commands. This may be caused by a high startup rate (two decades per minute in the source range or three decades per minute in the intermediate range) or by an asymmetric rod condition. These conditions can be considered as two separate OUT INHIBITS.

a. The 2 DPM SUR in source range or 3 DPM in intermediate range will impose an OUT INHIBIT in either AUTO or MANUAL.

b. If the reactor is at more than 60% power and either the safety rods are not OUT or a 9 inch Asymmetric Fault exists, an OUT INHIBIT will be imposed in AUTO.

Form ES-401-6

Written Exam Question Worksheet

Q # 066

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 016 K1.03 Page 3.7-9 Tier # 2

RO/SRO Importance Rating 3.2 3.2 Group # 2

Measurement Knowledge of the physical connections and/or cause-effect relationships between the NNIS and the following systems: Steam dump system

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

Which ONE (1) of the following identifies the instrumentation that provides the pressure input used for control of the listed components?

A. Turbine Bypass Valves - Turbine Header Pressure
Atmospheric Dump Valves - Turbine Header Pressure

B. Turbine Bypass Valves - Turbine Header Pressure
Atmospheric Dump Valves - OTSG Outlet Pressure

C. Turbine Bypass Valves - OTSG Outlet Pressure
Atmospheric Dump Valves - Turbine Header Pressure

D. Turbine Bypass Valves - OTSG Outlet Pressure
Atmospheric Dump Valves - OTSG Outlet Pressure

Technical Reference DWG D553730. Rev. R

Open Exam Reference

Learning Objective IV.E.09.01 Given a specific NNI detector, describe the following:
b. Interlocks or control functions provided by the detector.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .2 TO .9 ☐ 55.43 ☑ 55.45 .7, .8

Discriminant Validity Statements

A Plausible misconception
B Plausible misconception
C Plausible misconception
D Correct answer

Comments None.

Form ES-401-6

Written Exam Question Worksheet

Q # 067

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 026 K2.01 Page 3.5-10 Tier # 2

RO/SRO Importance Rating 3.4 3.6 Group # 2

Measurement Knowledge of bus power supplies to the following: Containment spray pumps

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** A

BS-P-1B is running for surveillance. Which ONE (1) condition would result in automatic trip of the BS-P-1B breaker?

A. 1A Aux Transformer fault with Auto Transfer of loads to 1B Aux Transformer.

B. 1B Aux Transformer fault with Auto Transfer of loads to 1A Aux Transformer.

C. Fault downstream of 1P 480v Bus low side feeder breaker.

D. Fault downstream of 1S 480v Bus low side feeder breaker.

Technical Reference OPM A-01 BOP and 1E Electrical Distribution, Rev 19 pages 36,37

Open Exam Reference

Learning Objective IV.A.15.08

Question Source ☐ New ☑ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☑ Used in Last Two TMI NRC Exams Exam Date 1998 SRO EXAM

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☐ 55.45

Discriminant Validity Statements

A Correct answer.

B Plausible misconception that automatic transfer occurs on 4KV ES Switchgear, coupled with incorrect power supply. The ES bus does not transfer to alternate Auxiliary Transformer, but rather to the backup emergency diesel.

C. Plausible misconception that 480 V Bus fault results in loss of 4KV feeder Bus coupled with incorrect power supply for BS-P-1B.

D Plausible misconception that 480 V Bus fault results in loss of 4KV feeder Bus.

Comments None.

FOR INFORMATION ONLY

SECTION A-01
REVISION 19

TMI OPERATIONS PLANT MANUAL

BOP AND 1E ELECTRICAL DISTRIBUTION (INCLUDING SUBSTATION)

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25	17	50	17	75	17	100	17

Concur

Owner — Randy S. Campbell — Date 10-15-99

Operations Training Manager, TMI — R H Carnella — Date 10/20/99

Training Coordinator, TMI — Linda L. Bitter — Date 10-25-99

Approved

Plant Operations Director TMI — James R Powers — Date 10/25/99

c. UV trips all equipment except block 1. Non ES equipment is locked out. Refer to Section 4 following. ES prevents auto transfer of 1C ES Valves CC.

d. Diesel Generator Breaker closing re-initiates block loading.

NOTE
If for some reason the ES system is bypassed after actuation, block loading will not occur again. Only block 1 equipment would restart on the diesel.

9.3.6 ES Simultaneous with Undervoltage

a. 27/86 locks out Non-ES equipment.

b. ES/Diesel breaker closing initiates block loading.

c. ES prevents auto transfer to 1C ES Valves CC.

9.3.7 Undervoltage Followed by ES

a. Undervoltage trips all equipment. Auto transfer of 1C ES Valves, panels PM1 and PM2, and fire pump occurs if alternate source is energized.

b. DG starts and energizes bus. Loads may be manually started.

c. ES with DG Breaker closed energizes 27/86. Non-ES equipment is tripped and locked out.

d. ES initiates block loading.

9.3.8 Equipment locked out on ES with UV

a. From D 4KV Bus

1. 1N 480V Bus Feeder

b. From 1P 480V Bus by 27/86

1. Tie breaker 1P-12, Trip and lock open

2. NS-P-1A (B), Trip and LO Non-ES Pump

c. From 1R 480V Bus by 27/86

1. Tie breaker 1R-12, Trip and lock open

2. SW-P-1A, Trip and LO

3. SR-P-1A, Trip and LO

4. NR-P-1A (B), Trip and LO Non-ES Pump

5. Feeder to Fire Pump ABT FS-P-2, Trip and LO

d. From E 4KV Bus

None

e. From 1S 480V Bus

1. Tie breaker 1S-12, Trip and LO

2. NS-P-1B (C), Trip and LO Non-ES Pump

f. From 1T 480V Bus

1. 1T-12 Tie breaker, Trip and LO

2. SW-P-1B, Trip and LO

3. SR-P-1B (C), Trip and LO

4. NR-P-1B (C), Trip and LO Non-ES Pump

5. Feeder to Fire Pump ABT FS-P-2, Trip and LO

9.3.9 Automatic Action of Equipment on Control Centers - 1A ES Control Center

> **NOTE**
>
> Equipment is divided into 3 sections: equipment that starts automatically on restoration of voltage, equipment that starts if in standby, equipment that must be manually started. LO indicates equipment that is locked out on UV with ES.

a. Starts automatically as needed.

			UV with ES
1.	Aux. Bldg. Sump (in auto)	WDL-P-5A	
2.	Instrument Air Comp. (can be started (from Control Room)	IA-P-1A	Prevents Auto-Start
3.	EFW Pump Room Sump	SD-P-3A	
4.	EFW Pump Room Sump	SD-P-4A	
5.	Makeup Pp Main Oil Pp	MU-P-3A	
6.	Control Twr Inst. Air Comp	AH-P-8A/B	
7.	Control Bldg. Booster Fan (starts when AH-E-18A/B starts)	AH-E-95A	

Form ES-401-6

Written Exam Question Worksheet

Q # 068

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 029 A2.03 Page 3.8-7 Tier # 2

RO/SRO Importance Rating 2.7 3.1 Group # 2

Measurement Ability to predict the impacts of the following malfunctions or operations on the Containment Purge System; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Startup operations and the associated required valve lineups

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

A RB purge was just initiated using AH-E-6A and AH-E-7A when AH-E-6A trips, resulting in the purge being secured automatically.

Which ONE (1) of the following actions is required in order to restore the purge?

Start AH-E-6B, then:

A. open AH-V-1C/D.

B. open AH-V-1A/B.

C. open AH-V-1A/B AND restart AH-E-7A.

D. open AH-V-1C/D AND restart AH-E-7A.

Technical Reference OP 1102-14, Reactor Building Purge and Venting, page 8, Rev 40.

Open Exam Reference

Learning Objective IV.F.02.06 Given any plant condition, identify Reactor Building Ventilation System functions and responses to high alarms concerning the following equipment:
I. AH-V-1A/1B/1C/1D RB purge supply and exhaust valves open/close controls

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☑ 55.43 .5 ☑ 55.45 .3, .13

Discriminant Validity Statements

A Correct answer.

B Plausible if interlock associated with AH-E-6A is misunderstood.

C Plausible if interlock associated with AH-E-6A is misunderstood.

D Plausible if interlock associated with AH-E-6A is misunderstood.

Comments None.

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Q-068

GPU Nuclear

TMI-1
Operating Procedure

Number: 1102-14

Title: Reactor Building Purge and Venting

Revision No.: 40

Applicability/Scope: TMI-1 Division

Responsible Office: Plant Ops. Dir.

Effective Date: 03/06/98

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

TRAINING DEPT.

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20.0	**40**						

	Signature	Date
Originator	George J Bajda	2/25/98
Originator	J Weatherling	2-25-98
Procedure Owner	[illegible]	3-2-98
PRG	[illegible]	3-3-98
Approver	L G Noll	3-3-98

1389c

b. Do not try to force the de-clutch lever from motor operation position to hand operation position. Move the de-clutch lever gently until engagement is felt.

c. Do not depress the de-clutch lever during motor operation to stop valve travel.

3. The Instrument Compressed Air System (Operating Procedure 1104-25) must be in operation to provide the necessary air supply for operation of the rotary valve actuators for valves AH-V-1A and AH-V-1D.

4. The Penetration Pressurization System (Operating Procedure 1104-21) must be in operation to maintain purge interspace pressure.

5. The following conditions will automatically close valves:

a. Purge Exhaust Valves AH-V-1A and AH-V-1B:

i) When fans AH-E-7A and AH-E-7B are shut down.

ii) High radiation signal as detected by radiation monitor RM-A9.

iii) Engineered Safeguard Signal. See paragraph 3.4 for operating sequence.

iv) Loss of control power.

NOTE

This condition only applies to valve AH-V-1A. Should a loss of control power occur when AH-V-1B is open, it will have to be manually closed.

v) Loss of instrument air for control of valve pneumatic actuators.

NOTE

This condition only applies to valve AH-V-1A.

b. Purge Supply Valves AH-V-1D and AH-V-1C:

i) When fans AH-E-6A and AH-E-6B are shut down or when fans AH-E-7A and AH-E-7B are shut down.

ii) High radiation signal as detected by radiation monitor RM-A9.

iii) Engineered Safeguard Signal. See paragraph 3.4 for operation sequence.

iv) Loss of control power.

Form ES-401-6

Written Exam Question Worksheet

Q # 069

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 033 A3.02 Page 3.8-10 Tier # 2

RO/SRO Importance Rating 2.9 3.1 Group # 2

Measurement Ability to monitor automatic operation of the Spent Fuel Pool Cooling System including: Spent fuel leak or rupture

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

Plant conditions:
- Reactor power is 100%.
- SF-P-1A Spent Fuel Cooling Pump is operating in recirc mode.
- SF-P-2 Borated Water Recirc Pump is operating for Pool A (LWDS) cleanup.
- Spent fuel assembly rack location change in progress in Spent Fuel Pool A.

Due to equipment failure, a spent fuel assembly is dropped – lying on top of the fuel racks in Pool A. Fuel pin clad rupture has resulted in significant activity release to the spent fuel pool water.

Identify the ONE (1) statement below that describes plant design response to RM-L-5 indication, which pegs high due to the increase in radioactivity described above.

A. SF-P-2 trips.

B. SF-P-1A trips.

C. MAP C-1-1 RADIATION LEVEL HIGH actuates.

D. MAP C-1-1 RADIATION LEVEL HIGH actuates AND Fuel Handling Building Supply Fan, AH-E-10, trips.

Technical Reference ARP MAP C-1-1, Radiation Level Hi, Rev. 20, Page 53.

Open Exam Reference

Learning Objective IV.E.06.04 Given an RMS channel, state the interlocks associated with a channel in the Radiation Monitoring system.

Question Source ☐ New ☑ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question # 69

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .5

Discriminant Validity Statements

A Plausible misconception that LWDS cleanup flow should be automatically terminated to minimize plant personnel radiation exposures due to contamination of LWDS cation demineralizer resins.

B Plausible misconception that liquid flow out of the spent fuel pool should be terminated to limit curie transport out of the spent fuel pool in order to minimize plant personnel radiation exposures.

C Correct answer.

D Plausible misconception that air flow out of the spent fuel handling area should be minimized to reduce plant personnel radiation exposures.

Comments None.

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GPU NUCLEAR	TMI Alarm Response Procedure	Number **MAP C**
Title **Main Annunciator Panel C**		Revision No. **72**
Applicability/Scope TMI Division	Responsible Office Plant Operations Dir.	Effective Date **09/07/99**

This document is within QA plan scope [X] Yes [ ] No
Safety Reviews Required [X] Yes [ ] No

TRAINING DEPT.

C-1-1 **Rev. 20** **09/07/99**	C-1-2 Rev. 12 02/27/98	C-1-3 Rev. 10 02/27/98	C-1-4 Rev. 9 02/27/98	C-1-5 Rev. 8 02/27/98	C-1-6 Rev. 7 02/27/98	C-1-7 Rev. 9 02/27/98	C-1-8 Rev. 11 10/23/98
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C-3-1 Rev. 10 02/27/98	C-3-2 Rev. 12 02/27/98	C-3-3 CANCELLED 11/01/91	C-3-4 CANCELLED 11/01/91	C-3-5 CANCELLED 11/01/91	C-3-6 CANCELLED 11/01/91	C-3-7 Rev. 7 02/27/98	C-3-8 Rev. 8 02/27/98

	Signature	Date
Originator	R. Bay	8-26-99
Procedure Owner	R. Bay	8-26-99
PRG	[illegible]	8/31/99
Approver	[illegible]	8/27/99

ALARM:

RM-L-5 (SPENT FUEL COOLING WATER SYSTEM)

SET POINTS:

Refer to Operating Procedure 1101-2.1, RMS setpoints

CAUSES:

Hi activity in the Spent Fuel Cooling System.

AUTOMATIC ACTION:

None

OBSERVATION (CONTROL ROOM):

1. RM-L-5 "Alert" alarm on PRF
2. RM-L-5 "Hi Alarm" on PRF
3. RM-L-5 Indication on PRF > above setpoints

MANUAL ACTION REQUIRED:

1. Refer to 1202-12, Excessive Radiation Levels.
2. Take a sample of SF cooling water at SF-V-72A or SF-V-72B. Have Rad Con perform analysis to verify alarm.
3. Clean up the SF cooling water system, using cation demineralizer in the Liquid Waste Disposal System in accordance with Operating Procedure 1104-29C, "Spent Fuel Cooling Water Cleanup Process".

Form ES-401-6

Written Exam Question Worksheet

Q # 070

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 035 K6.01 Page 3.4-15 Tier # 2

RO/SRO Importance Rating 3.2 3.6 Group # 2

Measurement Knowledge of the effect that a loss of the following will have on the S/Gs: MSIVs

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

The plant is at 40% power when the following events occur:
- A main steam line break occurs in the turbine building.
- The reactor is successfully tripped.
- Attempts to trip the turbine from the control room fail.
- Actions to isolate both OTSGs are initiated.
- MS-V-1A and MS-V-1C fail to close.
- All other OTSG isolation efforts are successful.

Which ONE (1) of the following conditions will occur?

A. OTSG 'A' blows down completely. OTSG 'B' is isolated.

B. OTSG 'B' blows down completely. OTSG 'A' is isolated.

C. OTSGs 'A' and 'B' blow down until the turbine is tripped locally.

D. OTSGs 'A' and 'B' blow down until generator output breakers trip on reverse power.

Technical Reference GAI Drawing 302-011 Rev 57.

Open Exam Reference

Learning Objective V.G.01.08 Explain how the plant responds to a double-ended steam line rupture per the TMI-1 FSAR.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .7

Discriminant Validity Statements

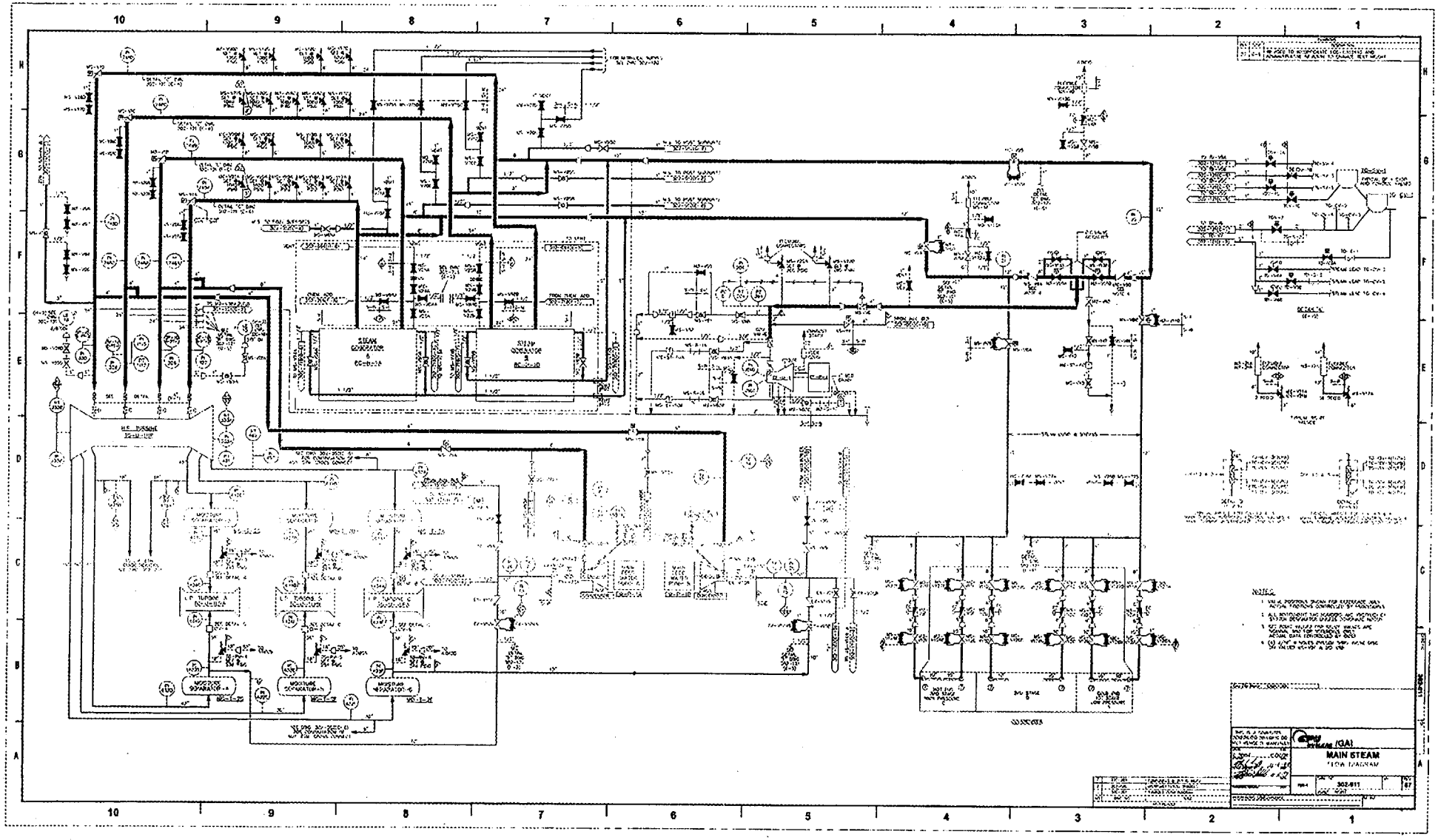
A Plausible if misunderstanding of steam flow path exists.

B Plausible if misunderstanding of steam flow path exists.

C Correct answer.

D Plausible if misunderstanding exists concerning status of turbine when generator output breakers trip.

Comments None.


MAIN STEAM
FLOW DIAGRAM

Form ES-401-6

Written Exam Question Worksheet

Q # 071

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 039 A3.02 Page 3.4-21 Tier # 2

RO/SRO Importance Rating 3.1 3.5 Group # 2

Measurement Ability to monitor automatic operation of the MRSS including: Isolation of the MRSS

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

Which ONE (1) of the following describes how Power Load Unbalance actuates the Combined Intercept Valves in order to prevent Main Turbine overspeed from the steam in the Moisture Separators?

A. The Intercept Valves fast close and are released within one second.

B. The Intercept Valves fast close and are released only when < 40% load mismatch.

C. The Intercept Stop Valves fast close and are released within one second.

D. The Intercept Stop Valves fast close and are released only when < 40% load mismatch.

Technical Reference OPM Section H-06, Electro-Hydraulic Control System, Rev. 12 pages 32 & 33

Open Exam Reference

Learning Objective IV.E.03.06 List 5 actions that occur when the EHC system power-load unbalance relay actuates.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☐ 55.43 ☑ 55.45 .5

Discriminant Validity Statements

A Correct answer.

B Plausible since PLU only clears at < 40% mismatch.

C Plausible misconception of the actuated valves.

D Plausible misconception of the actuated valves along with PLU clearing at < 40% mismatch.

Comments None.

FOR INFORMATION ONLY

SECTION H-06
REVISION 12

TMI OPERATIONS PLANT MANUAL

ELECTRO-HYDRAULIC CONTROL SYSTEM

Table of Effective Pages

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25	10	50	10	75	10		

Concur

Owner _____ Date 10/12/99

Operations Training Manager, TMI _____ Date 10/12/99

Training Coordinator, TMI _____ Date 10-12-99

Approved

Plant Operations Director TMI _____ Date 10/12/99

i. Both Feedwater Pumps Tripped

❶ Each Main Feedwater Pump has three hydraulic pressure switches which sense pressure to the FW pump turbine stop valve actuators. The FW pump is TRIPPED if pressure indicates less than 75 psig on 2/3 switches:

a) FW-PS-1484A, FW-PS-15A & FW-PS-16A - A PUMP

b) FW-PS-1484B, FW-PS-15B & FW-PS-16B - B PUMP

❷ A turbine Trip occurs if BOTH Feedwater pumps indicate TRIPPED.

j. Generator Faults

❶ (11 separate trip initiators) The turbine will trip on any one of the various generator fault signals. See Generator Section for more information on generator faults.

- 86 Lockout
- 86GS Generator Ground
- 186B Unit Differential
- 86NS Generator Negative Sequence
- 186T Main transformer Differential
- 86FA/86FB Generator Field Failure
- 86FP-M1A/86FP-M1B Main Transformer Fault
- 186TN Main Transformer Neutral
- 86P Generator Reverse Power

5.2.4.6 Overspeed Control Actions

a. Power Load Unbalance (PLU) (Figure 19)

PLU is designed to anticipate the increase in turbine speed following a significant (>40% load loss) load rejection, prevent a turbine trip and minimize the increase in turbine speed by fast closing the CVs & IVs.

❶ If Turbine load is 40% greater than Generator Load and Generator Load has changed at a rate greater than 2857%/sec. (equivalent of 100% load reduction in 35 msec.) then PLU is actuated after a 8 msec. time delay.

❷ Turbine load (6th stage pressure) and Generator load (total current) are both scaled in % of rated.

❸ When PLU actuates the following occurs:

a) Control Valve FASs are energized to dump close the Control Valves, and the valves remain closed until the 40% delta is cleared.

b) The Intercept Valve FASs are energized to dump close the IVs and then de-energize within 1 sec.

c) Load Reference signal is set to the No Load Reference (3%).

d) Load Set will runback at 150%/minute to a minimum of 3% or as long as the 40% delta is present.

e) ICS Turbine control H/A transfers to HAND and ICS goes to TRACK.

b. Intercept Valve Trigger (IVT) (Figure 20)

IVT is designed to minimize turbine speed increase following a load rejection of 20 - 40% of rated load.

❶ With IVT enabled, if IV-1 or IV-3 or IV-5 develop a position error, with valve position 10% above reference, IVT will actuate CLOSING ALL SIX IVs, and immediately release valves for Speed control.

❷ IVT is ENABLED when 6th stage pressure is >20 psig except during CIV 1,3 or 5 stroke testing.

c. Sequential Reopening

❶ If either PLU or IVT energizes FAS, valves do not de-energize immediately after the Logic releases control.

❷ There is a time delay on de-energizing which is designed to sequential release valves and minimize drop in EHC pressure caused by opening several valves at one time.

5.2.5 Testing Functions

5.2.5.1 Valve Testing (Figure 21I)

a. Stop Valve (SV) Stroke Testing

❶ Permissives

a) ALL other SV are OPEN
b) No limits or runbacks in effect
c) CLOSE VALVES not selected

❷ Operator initiates test after selecting FULL or PARTIAL stroke test mode.

❸ Valve stops at 60% OPEN and test terminates for Partial Test, then valve reopens.

❹ In Full Stroke mode, Fast Acting Solenoid (FAS) is energized when valve reaches 10% OPEN.

Form ES-401-6

Written Exam Question Worksheet

Q # 072

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 055 K1.06 Page 3.4-33 Tier # 2

RO/SRO Importance Rating 2.6 2.6 Group # 2

Measurement Knowledge of the physical connections and/or cause-effect relationships between the CARS and the following systems: Process radiation monitors

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

Plant conditions:
- Reactor tripped from 100% power.
- LOSS OF STATION POWER has occurred.
- Primary and secondary parameters are being maintained in the P-T Plot post trip stability windows.
- NO increase in RCS activity following trip.
- Mass balance calculations indicate the existence of 5 gpm RCS leakage.
- 1210-1 Immediate Actions are being performed.

Considering the conditions above, identify the ONE (1) instrument below that, if evaluated alone, would support a valid determination that OTSG tube leakage exists.

A. RM-A-2 (RB Atmospheric Monitor).

B. RM-A-5 (Condenser Off-Gas Monitor).

C. RM-G-25 (Condenser Off-Gas Monitor).

D. RM-G-26 (Main Steam Line Monitor).

Technical Reference 6610-PLN-4200.02, TMI Emergency Dose Calculation Manual (EDCM), Rev. 11, Page 51.

Open Exam Reference

Learning Objective V.E.11.02 Concerning the ATOG Symptom & OTSG Tube leakage condition:
- State the "Symptoms" in order of priority
- Explain the concerns associated with each symptom
- Describe the relative priority of the symptoms

Question Source ☐ New ☑ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question # 72

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .2 to .9 ☐ 55.43 ☑ 55.45 .7 to .8

Discriminant Validity Statements

A Plausible misconception since this monitor would indicate the existence or absence of RCS leakage into the RB, but does not in itself exclude the existence of OTSG tube leakage.

B Plausible misconception because this monitor would indicate the existence of OTSG tube leakage -- in the presence of condenser off-gas flow.

C Plausible misconception because this monitor would indicate the existence of OTSG tube leakage -- in the presence of condenser off-gas flow.

D Correct answer because it monitors main steam line, not condenser off-gas activity.

Comments Answer reference was updated from Rev. 10 to Rev. 11 with no impact on answer - 2/8/00.

AmerGen	TMI - Unit 1 Radiological Controls Procedure	Number 6610-PLN-4200.02
Title TMI Emergency Dose Calculation Manual (EDCM)		Revision No. 11
Applicability/Scope TMI Division	Responsible Office Mgr., Rad Engrng.	Effective Date 01/28/00

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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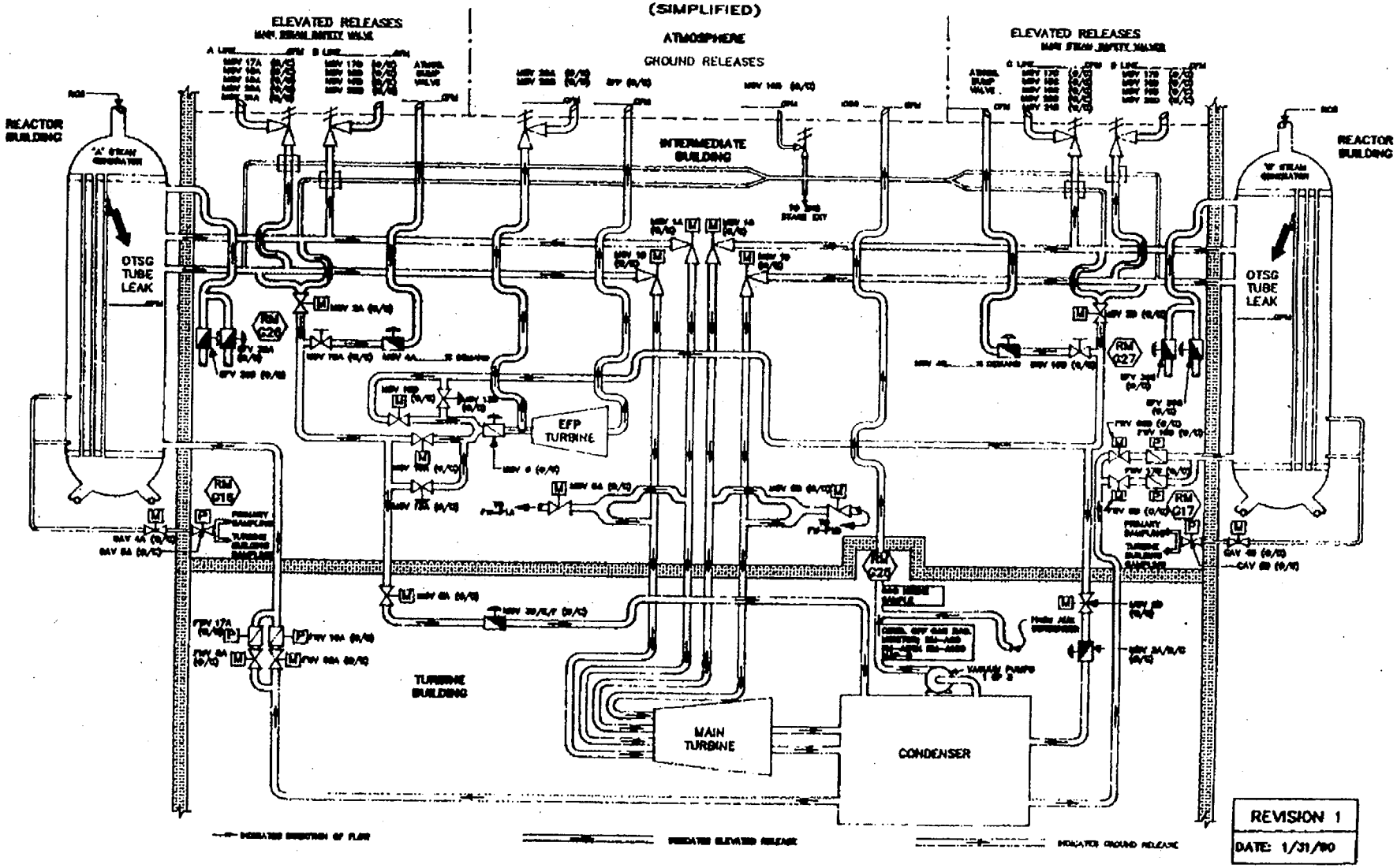
	Signature	Date
Originator	[illegible]	1/20/2000
Procedure Owner	[illegible]	1/20/2000
PRG	[illegible]	1/27/2000
Approver	Jim Felfer	1/26/00

FIGURE 5.9-3

6610-PLN-4200.02
Revision 11

OTSG LEAK/RUPTURE RELEASE PATHWAYS
(SIMPLIFIED)



Form ES-401-6

Written Exam Question Worksheet

Q # 073

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 062 A2.08 Page 3.6-4 Tier # 2

RO/SRO Importance Rating 2.7 3.0 Group # 2

Measurement Ability to predict the impacts of the following malfunctions or operations on the AC Distribution System; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Consequences of exceeding voltage limitations

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** B

The plant is experiencing low system grid voltage.

Which ONE of the following conditions occurring on the 1D 4160V Bus will cause the feeder breaker to trip and the emergency diesel to start and load the bus?

A. 3800V for 20 seconds

B. 3200V for 12 seconds

C. 2600V for 6 seconds

D. 2000V for 1 second

Technical Reference AbP 1203-41, Low System (Grid) Voltage, Rev. 21 page 2

Open Exam Reference

Learning Objective IV.G.06.07 Describe major system interlocks and protective features for the Class 1E electrical system.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☑ 55.43 .5 ☑ 55.45 .3, .13

Discriminant Validity Statements

A Plausible, but must be < 3760V for > 10 seconds

B Correct answer. < 3760V for > 10 seconds

C Plausible, but must be < 2400V for > 1.5 seconds

D Plausible, but must be < 2400V for > 1.5 seconds

Comments None.

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AmerGen

TMI - Unit 1
Abnormal Procedure

Number: **1203-41**

Revision No.: **21**

.le

Low System (Grid) Voltage

Applicability/Scope	Responsible Office	Effective Date
TMI Division	Plant Operations Dir.	01/14/00

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

SIMULATOR ROOM FILE COPY

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	Signature	Date
Originator	Earl Showalter	1/13/2000
Procedure Owner		1/13/00
PRG		1-14-2000
prover DWS		1/14/00

1.0 SYMPTOMS

1.1 Sustained low voltage condition, less than 232.4KV indicated on either:

a. Substation voltage Panel SS-1

b. Computer Alarm "4 or 8 Bus Voltage Lo/Hi" (Pt. A0946)

1.2 Possible "4KV BOP Bus Trouble" alarm (AA-3-3).

1.3 Possible "4KV ES Bus Trouble" alarm (B-3-1).

1.4 Possible "4KV BOP Bus UV" alarm (AA-2-3).

1.5 Possible "4KV ES Bus UV/OV" alarm (B-2-1).

1.6 Possible "480V BOP Bus UV" alarm (AA-2-5).

1.7 Possible "480V ES Bus UV/OV" alarm (B-2-4).

1.8 Possible motor overload alarms or high stator temperature indication.

2.0 IMMEDIATE ACTION

2.1 Automatic Action

> **NOTE**
>
> A degraded grid condition with LOCA can result in separation between ES loads and offsite power. **For a single auxiliary transformer inservice,** this can occur at 232.4 KV. **For both auxiliary transformers inservice,** separation can occur at 224 KV.

2.1.1 Motors may trip on under voltage or overcurrent.

2.1.2 D and/or E Bus feeder breaker will trip, associated 4KV and 480V Bus motor breakers will trip, and diesel will start and energize the bus if voltage is ≤ 3760 volts for > than 10 seconds; or ≤ 2400 volts for > 1.5 seconds.

2.1.3 Main Generator VARS may increase if voltage control is in automatic, and VAR limit may be approached or exceeded.

Form ES-401-6

Written Exam Question Worksheet

Q # 074

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 063 K3.02 Page 3.6-6 Tier # 2

RO/SRO Importance Rating 3.5 3.7 Group # 2

Measurement Knowledge of the effect that a loss or malfunction of the DC Electrical System will have on the following: Components using DC control power

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** A

Initial plant conditions:
- Reactor power was 100%.

Sequence of events:
- Loss of entire 1A 125-250 Volt DC Electrical Distribution System.
- Manual Reactor Trip due to Main Feedwater control problems.
- Loss of RCS Subcooled Margin.

From the list below, identify the ONE (1) statement that describes how to secure the RCPs under these conditions.

A. Trip ALL 4 RCPs using their associated extension controls (4) on Console Center.

B. Trip RC-P-1A and RC-P-1C using extension controls on Console Center;
Stop RC-P-1B and RC-P-1D by de-energizing 1B 6900 Volt Buses at Panel PR.

C. Trip RC-P-1B and RC-P-1D using extension controls on Console Center;
Stop RC-P-1A and RC-P-1C by de-energizing 1A 6900 Volt Buses at Panel PR.

D. Stop ALL 4 RCPs by de-energizing 1A and 1B 6900 Volt Buses.

Technical Reference EP 1202-9A, Loss of A DC Distribution System, Rev. 41, Page 3.

Open Exam Reference

Learning Objective V.E.02.02 State and explain the Immediate Action(s) of ATP 1210-2, Loss of 25f Subcooled Margin.

Question Source ☐ New ☑ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question # 13

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .6

Discriminant Validity Statements

A Correct answer.

B Plausible distracter if control power scheme for RCPs is misunderstood.

C Plausible distracter if control power scheme for RCPs is misunderstood.

D Plausible distracter if control power scheme for RCPs is misunderstood.

Comments None.

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GPU NUCLEAR	TMI Emergency Procedure	Number 1202-9A

Title	Revision No
Loss of "A" DC Distribution System	41

Applicability/Scope	Responsible Office	Effective Date
TMI Division	Plant Operations Dir.	08/19/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	Brantley for Doug Gause	8-6-99
Procedure Owner	Stuart Brantley	8-6-99
PRG	[illegible]	8/12/99
Approver	James Naulls	8/10/99

7. Suction Valves to VA-P-1A and VA-P-1C will close.

8. Emergency Feed Pump EF-P-1 Starts.

9. Emergency Diesel Generator EG-Y-1A starts

10. Panel 1M automatically transfers to "B" DC Distribution System.

11. RC-RV-2 fails closed and is inoperable

12. MU-V-6A fails open and MU-V-6B fails closed causing "A" makeup and purification demineralizer to be in service.

13. MU-V-11B fails open and MU-V-11A fails closed causing MU-F-1B to be in service.

14. CM-V-1 and CM-V-3 fail closed causing RM-A-2 to become inoperable.

15. Loss of Control Power to the following:

    a. 1A and 1B 6900V Reactor Plant Swgr. Feeder Breakers

    b. 1D-4160V ES Swgr.

    c. 1A-4160V Turbine Plant Swgr.

    d. 1C-4160V Turbine Plant Swgr.

    e. 1C-480V Turbine Plant Swgr.

    f. 1E-480V Reactor Bldg. Ht. and Vent Swgr.

    g. 1G-480V Reactor Plant Swgr.

    h. 1H-480V Aux. and Fuel Bldg. Ht. and Vent Swgr.

    i. 1J-480V Turbine Plant Swgr.

    j. 1N-480V Turbine, Reactor and Control Bldgs. Htg. Swgr.

    k. 1P-480V ES Swgr.

    l. 1R-480V Scrn. Hse. ES Swgr.

NOTE
The unit should continue to operate unless tripped by a loss of main condenser vacuum or high RCS pressure.

Form ES-401-6

Written Exam Question Worksheet

Q # 075

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 064 A2.03 Page 3.6-10 Tier # 2

RO/SRO Importance Rating 3.1 3.1 Group # 2

Measurement Ability to predict the impacts of the following malfunctions or operations on the ED/G system; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Parallel operation of ED/Gs

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

EG-Y-1B is in parallel operation with the 1E 4160V bus for a surveillance with the following conditions:
- Exciter switch is in Manual
- 1E Bus Voltage is 4165V
- 1E Bus Frequency is 60 HZ
- Diesel load is 2.9MW
- MVARs is 2.2 MVAR

A perturbation on the 8 bus results in the 1E bus voltage reducing to 4120V.

Which ONE of the following operator actions would be required to compensate for this event?

A. Raise diesel speed using EG-Y-1B Governor Control Switch

B. Lower diesel speed using EG-Y-1B Governor Control Switch

C. Raise diesel output voltage using the Manual Voltage Controller

D. Lower diesel output voltage using the Manual Voltage Controller

Technical Reference OP 1107-3, Diesel Generator, Rev. 90 page 23

Open Exam Reference

Learning Objective IV.G.08.05 List the normal operating parameters on Control Room indications for the Emergency Diesel Generators, and the action required if these parameters are not in allowable band.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☑ 55.43 .5 ☑ 55.45 .3, .13

Discriminant Validity Statements

A Plausible if misunderstanding of speed change effects on diesel MVAR loading exists.

B Plausible if misunderstanding of speed change effects on diesel MVAR loading exists.

C Plausible if required change in output voltage to reduce MVARS is misunderstood.

D Correct answer.

Comments None.

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GPU NUCLEAR

TMI
Operating Procedure

Number: 1107-3

Title: **Diesel Generator**

Revision No.: 90

Applicability/Scope: TMI Division

Responsible Office: Plant Operations Dir.

Effective Date: 08/10/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

TRAINING DEPT.

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	Signature	Date
Originator	D James	8-2-99
Procedure Owner	R E Bo	8-3-99
PRG	[illegible]	8/5/99
Approver	James Paules	8/4/99

_____ 15. Using the EG-Y-1A (1B) Governor Control Switch, quickly put load on the diesel generator to prevent a reverse power trip of the generator output breaker G1-02 (G11-02). Initial load should be approximately 0.5 MW with 0 to 2.5 MVAR.

CAUTION

Operation of the Diesel Generator in parallel with the system (Grid) degrades the D/G reliability since system voltage and frequency fluctuations can cause an unwanted D/G trip (i.e., reverse power).

_____ 16. De-energize the synchroscope.

CAUTION

If, while loading the diesel generator, output current exceeds 440 amps, stop loading the diesel generator and evaluate all other diesel parameters (i.e.: MVAR's fuel rack position, etc....) to ensure the diesel generator is not run in an overload condition (> 3.0 MW). If necessary, Plant Engineering can aid in this evaluation.

NOTE

Enclosure 8 provides reference values for the diesel generator load indicating parameters for various engine loads. These values may be referred to for evaluation of engine operation.

_____ 17. Gradually load the diesel to the desired load using the GOVERNOR control switch (normally 2.9 ± 0.1 MW) (within 5-10 minutes).

a. Monitor MVARS and adjust the voltage control if necessary, to maintain 0 to 2.5 MVARS.

b. Evaluate diesel operating parameters; Amps, MVARS and Megawatts. Refer to Enclosure 8 as required.

NOTE

Until the governor is warm (30 to 40 minutes) the load on the diesel may tend to drift down.

_____ 18. Verify 2.9 ± 0.1 MW output

_____ 19. Verify approximately 0.5 to 1.0 MVar out Maximum of 2.25 MVar at 3.0 MW-refer to capability curve, Figure 1. Adjust MVARS, if required, using the voltage control on the electric equipment panel (generator MANUAL VOLTAGE CONTROLLER on console if EXCITER switch is in MANUAL).

Form ES-401-6

Written Exam Question Worksheet

Q # 076

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 073 K4.02 Page 3.7-15 Tier # 2

RO/SRO Importance Rating 3.3 3.9 Group # 2

Measurement Knowledge of PRM system design features and/or interlocks provide for the following: Letdown isolation on high RCS activity

Proposed Question ☑ RO ☐ SRO ☐ PRA Related **Correct Answer** C

Plant conditions:
- Reactor power is 100%.
- Significant increase in RCS activity causes high alarms on RM-L-1 and RM-L-1 Low.

Identify the ONE (1) statement below that describes the expected automatic action(s) for these plant conditions.

A. MU-V-3 closes.

B. MU-V-1A AND MU-V-1B close.

C. MU-V-2A AND MU-V-2B close.

D. MU-V-2A, MU-V-2B, AND MU-V-3 close.

Technical Reference EP 1202-11, High Activity in Reactor Coolant, Rev. 25, Page 2.

Open Exam Reference

Learning Objective IV.A.09.11 State the interlocks associated with the following MU&P system components:
b. MU-V-2A/2B

Question Source ☐ New ☑ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question # 14

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☐ 55.45

Discriminant Validity Statements

A Plausible because MU-V-3 closes on high letdown temperature.

B Plausible because interlock isolates RCS letdown.

C Correct answer.

D Plausible because MU-V-2A, MU-V-2B, and MU-V-3 close on high-high letdown temperature.

Comments None.

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GPU NUCLEAR

TMI
Emergency Procedure

Number
1202-11

Title
High Activity in Reactor Coolant

Revision No.
25

Applicability/Scope
TMI Division

Responsible Office
Plant Operations Director

Effective Date
12/15/98

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

TRAINING DEPT.

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Originator	[illegible]	12-2-98
Procedure Owner	L.G. Noll For T.A. Goodhouse	12-10-98
PRG	[illegible]	12-10-98
Approver	L.G. Noll	12-10-98

Title

High Activity in Reactor Coolant

1.0 **SYMPTOMS**

1. Alert Alarm on RC letdown radiation monitor RM-L1.

2. High Radiochemical analysis results (≥ 0.35 μCi/gm Dose Equivalent I-131 or > 100/E μCi/gm).

2.0 **IMMEDIATE ACTION**

A. Automatic Action

1. MU-V2A and MU-V2B close If a high alarm is received on RM-L1.

B. Manual Action

1. Announce on Page: "High Activity has been detected in letdown. All unnecessary personnel in the controlled area report to H.P. Checkpoint."

3.0 **FOLLOWUP ACTION**

Objective:

Determine activity in RCS, minimize radiation levels in the plant, minimize radioactive exposure to personnel, and to determine the cause for the high activity.

_____ 1. If RM-L1 is in high alarm, compare its reading to RM-L1 low. RM-L1 low should also have a high alarm.

If RM-L1 low is not tracking RM-L1.

Then have the Shift Supervisor evaluate restoring letdown per 1210.10.

_____ 2. If RM-L1 low is tracking RM-L1.

Then verify MU-V2A and MU-V2B are closed or close manually.

_____ 3. If letdown cannot be restored, then commence plant shutdown and cooldown to limit pressurizer level to ≤ 330 inches.

_____ 4. If pressurizer level reaches 380 inches while the reactor is critical, then trip the reactor.

_____ 5. Monitor RM-A5 channel for possible increase, if RM-A5 alert alarms and OTSG tube leak is suspected, refer to ATP 1210-5.

Form ES-401-6

Written Exam Question Worksheet

Q # 077

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 075 GEN. 2.4.2 Page 2-11 Tier # 2

RO/SRO Importance Rating 3.9 4.1 Group # 2

Measurement Knowledge of system setpoints, interlocks and automatic actions associated with EOP entry conditions

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

The following plant conditions exist:
- a Circulating Water tube leak has occurred on the "B" side of the condenser
- CE-6A (corrected feedwater cation conductivity) reads 6.5 umho/cm
- CE-6 (feedwater cation conductivity) reads 7.5 umho/cm
- Chemistry has validated the readings

Which ONE (1) of the following actions is required for these plant conditions?

A. Secure all Moisture Separator Drain Pumps and continue power operation.

B. Reduce power to less that 50% and isolate the "B" side Circulating Water loop.

C. Perform a normal plant shutdown and cooldown to Decay Heat Operations.

D. Trip the reactor and go to ATP 1210-1, Reactor Trip.

Technical Reference AbP 1203-5, High Contaminants in the Condensate and/or Feedwater System, Rev. 23 page 2

Open Exam Reference

Learning Objective V.C.02.03 Given a list of symptoms, identify the event and determine which Immediate Manual Actions are required IAW AP 1203-5.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .7, .8

Discriminant Validity Statements

A Plausible because this would stop recirculation of the contaminant.

B Plausible because this is the action required for lower contamination levels.

C Plausible if the examinee does not realize the immediacy of the problem.

D Correct answer.

Comments None.

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GPU NUCLEAR

TMI
Abnormal Procedure

Number: 1203-5

Title: High Contaminants in the Condensate and/or Feedwater System

Revision No.: 23

Applicability/Scope: TMI Division

Responsible Office: Plant Operations Dir.

Effective Date: 04/29/98

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

SIMULATOR ROOM
(Shift Supvs. Desk)

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	Signature	Date
Originator	J L Masters	4/16/98
Procedure Owner	R C Boyer	4-16-98
PRG	[illegible]	4-23-98
Approver	L. G. Noll	4-17-98

1.0 SYMPTOMS

1. The Control Room conductivity recorder indicates an increasing conductivity.

2. Increasing conductivity indicated on the recorders in the secondary sampling room, as well as on the sodium monitor at the condensate pump discharge.

3. Alarm PLB-8-6, "Conductivity Recorder Abnormal".

4. Alarm PRF 6-1, "CE-6A Conductivity Hi".

5. Alarm PRF 2-6, "CO-C1-A Conductivity Trouble".

6. Alarm PRF 2-7, "CO-C1-B Cndtvty Trouble".

7. Alarm PRF 2-8, "CO-C2A/B Cndtvty Trouble".

8. Alarm PLB-5-7, "Turbine Sampling Room Trouble", caused by increasing sodium in the OTSG Feedwater and in the Condensate System. (OTSG Feedwater over 3 ppb and increasing.)

9. Alarm on the following PPC (Plant Process Computer) points:

    1. A1031 - output from instrument CE-773 (normally monitoring CE-3, Condensate Pump Discharge)

    2. A1032 - output from instrument CE-772 (normally monitoring CE-2, Condensate Pump Outlet but is also used to monitor Ecolochem Makeup Water)

    3. A1033 - output from instrument CE-801 (normally monitoring CE-6, OTSG Feedwater)

2.0 IMMEDIATE ACTION

A. Automatic - None

B. Manual

1. Contact Chemistry Dept. to confirm abnormal conductivity with the cation conductivity recorders in the secondary sample room (i.e., an abnormally high cation conductivity will be indicated on more than one sample point).

2. If CE-6A (corrected feedwater cation conductivity) is confirmed to be greater than 5.0 µmho/cm or CE-6 (feedwater cation conductivity) is confirmed to be > 6.0µmho/cm, immediately trip the reactor and go to ATP 1210-1, Reactor Trip.

3. For elevated sodium and for cation conductivities below the above limits, refer to the follow-up actions.

Form ES-401-6

Written Exam Question Worksheet

Q # 078

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 079 A4.01 Page 3.8-22 Tier # 2

RO/SRO Importance Rating 2.7 2.7 Group # 2

Measurement Ability to manually operate and/or monitor in the control room: Cross-tie valves with IAS

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** A

The following plant conditions exist:
- A rupture occurs in the air header supplying MS-V-3s.
- IA-P-4 trips on overload.
- IA-P1A/B and SA-P-1A/B are started successfully.
- Instrument air primary (PI-222) and secondary (PI-1403) pressure indicators on PL are tracking together and continue to decrease.

Which ONE(1) of the following indications would be observed on PL when IA-V-26 closes?

When header pressure drops below:

A. 60 psig, PI-222 starts to increase and PI-1403 continues to decrease.

B. 60 psig, PI-1403 starts to increase and PI-222 continues to decrease.

C. 80 psig, PI-222 starts to increase and PI-1403 continues to decrease.

D. 80 psig, PI-1403 starts to increase and PI-222 continues to decrease.

Technical Reference EP 1202-36, Loss of Instrument Air, page 2, Rev.29.

Open Exam Reference

Learning Objective V.D.17.03 Given a list of symptoms, identify the event, determine what Immediate Automatic Actions will occur and what Immediate Manual Actions are required IAW EP 1202-36.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .5 TO .8

Discriminant Validity Statements

A Correct answer.

B Plausible if location of air leak is misunderstood.

C Plausible if wrong setpoint is interpreted.

D Plausible if wrong setpoint is interpreted and location of air leak is misunderstood.

Comments None.

FOR INFORMATION ONLY 078

GPU NUCLEAR	TMI Emergency Procedure	Number 1202-36
Title: **Loss of Instrument Air**		Revision No. 29
Applicability/Scope: TMI Division	Responsible Office: Plant Ops. Director	Effective Date: 06/22/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	David B. Wise	6/15/99
Procedure Owner	James N Paules	6/15/99
PRG	John S. LaRoche	6/15/99
prover	James N Paules	6/15/99

NOTE

The Instrument and Service Air Systems are normally Cross connected at discharge of IA-P-4. On a loss of IA-P-4 the IA and SA will be isolated from each other until IA-V-1 opens on low IA pressure.

1.0 SYMPTOMS

1. "Instrument Air Press Low Turbine Area" (PLB-1-7), "Instrument Air Press Low Aux Bldg Area" (PLB-2-7) and "Station Service Air Press Low" (PLB-1-8) annunciator alarms sound on PL panel annunciator "B".

2. Loss of running red indicator lights for IA-P-1A and B, SA-P-1A and B, IA-P-4 on panel "PL".

3. Low readings at instrument air (Primary Plant and/or Secondary Plant) and service air pressure indicators on PL panel.

4. Loss of control or erratic behavior of air operated component.

0 IMMEDIATE ACTION

A. Automatic Action

1. IA-V-1 (Instrument air backup supply valve from service air) opens at 80 $^{+0}_{-5}$ psig.

2. IA-V-26 closes at <60 psig to isolate Turbine Building and Intermediate Building headers.

3. Backup Instrument Air Compressor (IA-P-2A/B) will supply air to the critical component found on Table 2.

4. Turbine driven emergency feedwater pump (EF-P-1) starts due to steam inlet valves MS-V-13A (B) failing open.

5. Reactor coolant pumps trip (loss of seal injection and intermediate closed cooling water) approximately 10 minutes following Panel Left Alarm PLA-6-7 "Low Air ICV3, 4, 6, MUV20, 26".

6. Potential Reactor trip/Turbine trip.

7. Loss of penetration cooling (due to damper closure).

8. Condenser vacuum may be lost due to failing shut of Vacuum Pump suction valves VA-V-5's.

B. Manual Action

1. Make Plant Page and Radio announcement "Attention all personnel, a loss of Instrument Air is in progress. All personnel using Instrument or Service Air should stop immediately. Report any air leaks to Unit 1 Control Room."

Form ES-401-6

Written Exam Question Worksheet

Q # 079

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 086 A4.01 Page 3.8-24 Tier # 2

RO/SRO Importance Rating 3.3 3.3 Group # 2

Measurement Ability to manually operate and/or monitor in the control room: Fire water pumps

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** D

The Fire Protection system is in normal alignment when a leak occurs in the yard fire main. Fire main pressure drops to 85 psig and stabilizes.

Assuming NO operator action, identify the ONE (1) set of fire pumps that would be running at this time.

A. Fire Service Jockey Pump (FS-P-4)
Screen House Diesel Driven Fire Pump (FS-P-3)

B. Circulation Water Diesel Driven Fire Pump (FS-P-1)
Screen House Motor Driven Fire Pump (FS-P-2)

C. Screen House Motor Driven Fire Pump (FS-P-2)
Screen House Diesel Driven Fire Pump (FS-P-3)

D. Screen House Motor Driven Fire Pump (FS-P-2)
Fire Service Jockey Pump (FS-P-4)

Technical Reference OPM Section K-01, Fire Protection/Fire Service, Page 11 and 26, Rev 16.

Open Exam Reference

Learning Objective IV.D.35.06 Describe the four (4) operational modes of FS-P-1, FS-P-2 and 3.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 ,5 TO .8

Discriminant Validity Statements

A Plausible if auto start setpoints of fire pumps is misunderstood.

B Plausible if auto start setpoints of fire pumps is misunderstood.

C Plausible if auto start setpoints of fire pumps is misunderstood.

D Correct answer.

Comments None.

FOR INFORMATION ONLY

SECTION K-01
REVISION 16

TMI OPERATIONS PLANT MANUAL
FIRE PROTECTION/FIRE SERVICE

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Owner R E Siegley Date 1-11-00

Operations Training Manager, TMI R H Lamoglia Date 1/13/00

Training Coordinator, TMI Linda R. Ritter Date 1-14-2000

Approved

Plant Operations Director TMI D L Rich Date 1/14/00

4.3.2 Circulating Water Diesel Driven Fire Pump (FS-P-1, Unit-1)

a. Pressure indicator: PI-192, local indication of pump suction.

b. Pressure indicator: PI-153, local indication of pump discharge.

c. Pressure switch: PS-154, auto-start of diesel at system pressure of 70 psig.

d. Diesel Engine Indication of performance, high water temperature, low oil pressure, engine fail to start, engine running and engine overspeed alarms in Control Room and on computer data logger.

e. Level indicator: LI-214, diesel fuel oil tank level gauge.

4.3.3 Screen House Motor Driven Fire Pump (FS-P-2)

a. Pressure indicator: PI-193, local indication of pump discharge.

b. Pressure switch: PS-220, auto-start of diesel at system pressure of 90 psig.

4.3.4 Screen House Diesel Driven Fire Pump (FS-P-3)

a. Pressure indicator: PI-194, local indication of pump discharge.

b. Pressure switch: PS-221, auto-start of diesel at system pressure of 80 psig.

c. Diesel Engine Indication of performance, high water temperature, low oil pressure, engine fail to start, engine running and engine overspeed alarms in Control Room and on computer data logger.

d. Level indicator: LI-213, diesel fuel oil tank level gauge.

4.3.5 Wet Pipe Sprinkler Systems

A pressure switch is installed on each sprinkler alarm valve. The pressure switch is used to alarm locally and in the Control Room if the system is actuated by an open sprinkler head. For "Outbuildings" local alarms (above gongs) and remote alarms actuate in the Process Center at Panel PC-F. Each main isolation valve for the sprinkler alarm valve has a tamper switch installed that will alarm locally and in the Control Room if the isolation valve is closed more than two turns. Outbuilding Tamper Switch alarms are local and at the Process Center (PC-F).

4.3.6 Manual Preaction System

Each manual preaction system has heat activated devices (HAD's) which will alarm locally and in the Control Room. The piping has a supervisory air system which monitors the closed sprinkler heads. This system will also alarm in the Control Room.

4.2.5 Fire Service Jockey Pump (FS-P-4)

The Fire Service Jockey Pump FS-P-4 is a hydro pumping unit with a cut-in point of 108 psig and a cutout point of 125 psig to maintain a 108 to 125 psig pressure in the yard fire main. It is located along the east wall in the Turbine Building 305' elevation. The 25 gpm pump takes suction from the six (6) inch filtered water line from the altitude tank to the Plant for makeup. It discharges into the Turbine Building East Header/fire main. There are two (2) modes of operation provided for the fire service jockey pump by use of the HAND-OFF-AUTO switch. 1) Automatic Mode: pump will auto-start at system pressure of 108 psig and stop at system pressure of 125 psig as detected by pressure switch PS-586. 2) Hand Mode: the pump is started by depressing and releasing the start pushbutton. The pump remains in operation until the HAND-OFF-AUTO switch is placed in the OFF position.

Power is provided to the Jockey pump from the 480V IC ES Valve Control Center Breaker in Unit 9AR.

4.2.6 FS-P-1 and FS-P-3 Pump Controller

The fire pump controller for FS-P-1 and FS-P-3 is a Master Control Systems model DC-235 controller. The controller provides for three different modes of operation.

a. **Test Position**: Placing the control switch in the TEST position will start the engine immediately through the automatic control circuits in the same manner as when the engine starts with the selector switch in the AUTO position. The test position should be used for routine starting. This mode of operation brings into use all of the automatic cranking circuits, safety alarms and control room alarms.

b. **Automatic Position**: Placing the selector switch in the AUTO position energizes both blue "ON" pilot lamps. This indicates that the controller is in a standby position, ready to start the diesel when required to do so.

   The two blue "ON" pilot lamps are indicating that both battery banks are in service. Overload circuit breakers are located next to the blue pilot lamps. Any condition causing a tripping of the breaker will cause the corresponding blue lamp to go out. This indicates that the battery bank is not in service. When the trouble condition is corrected, the overload circuit breaker is depressed to reset. The blue pilot lamp will light up indicating that the battery bank is in service.

c. **Manual Positions**: Two manual positions are provided on the selector switch for starting the engine through the manual start pushbutton. When the switch is in MAN 1, diesel is started from battery bank 1 and in MAN 2, the diesel is started from battery bank 2. When the diesel is running with the selector switch in either manual position, the engine safety protection alarm circuits, except overspeed shutdown circuits, are not in operation.

d. **Off Position**: This is a safety position which prevents the diesel from starting during periods of maintenance. It is also used to reset the controller subsequent to a failure to start, overcrank, or overspeed trip.

Form ES-401-6

Written Exam Question Worksheet

Q # 080

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 005 A4.04 Page 3.4-13 Tier # 2

RO/SRO Importance Rating 3.1 2.9 Group # 3

Measurement Ability to manually operate and/or monitor in the control room: Controls and indication for closed cooling water pumps

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

Which ONE (1) of the following would be a positive indication of a successful start of DC-P-1A?

A. Panel PCR ESAS Status indication for DC-P-1A.

B. DC-P-1A ammeter indication on Console Center.

C. Red indicating light for DC-P-1A on Console Center.

D. DC-P-1A breaker status from the Plant Process Computer.

Technical Reference NA

Open Exam Reference

Learning Objective V.J.05.01 List 3 ways to monitor system and equipment parameters and operation.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .5 TO .8

Discriminant Validity Statements

A Plausible, but provides breaker status only and assumes 480V power is available.

B Correct answer.

C Plausible, but provides breaker status only and assumes 480V power is available.

D Plausible, but provides breaker status only and assumes 480V power is available.

Comments None.

Form ES-401-6

Written Exam Question Worksheet

Q # 081

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 007 K1.03 Page 3.5-2 Tier # 2

RO/SRO Importance Rating 3.0 3.2 Group # 3

Measurement Knowledge of the physical connections and/or cause-effect relationships between the PRTS and the following systems: RCS

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

The plant is at 100% power with normal system alignments.

Which ONE (1) of the following would result in a discharge of Reactor Coolant to the Reactor Coolant Drain Tank?

A. "A" Loop RCS Wide Range T-cold failing low.

B. Leakage through RV Head vent valves RC-V-42 & 43.

C. Selected RCS Wide Range Pressure failing high without SASS actuation.

D. Closure of RCP seal return valves MU-V-25 & 26 on an inadvertant 30# ESAS.

Technical Reference DWG 302-660 Rev. 38.

Open Exam Reference

Learning Objective IV.A.01.02 Draw and explain a simplified one line diagram of the Reactor Coolant System, showing all major components, valves and connections to other systems.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge

☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .2 TO .9 ☐ 55.43 ☑ 55.45 .7, .8

Discriminant Validity Statements

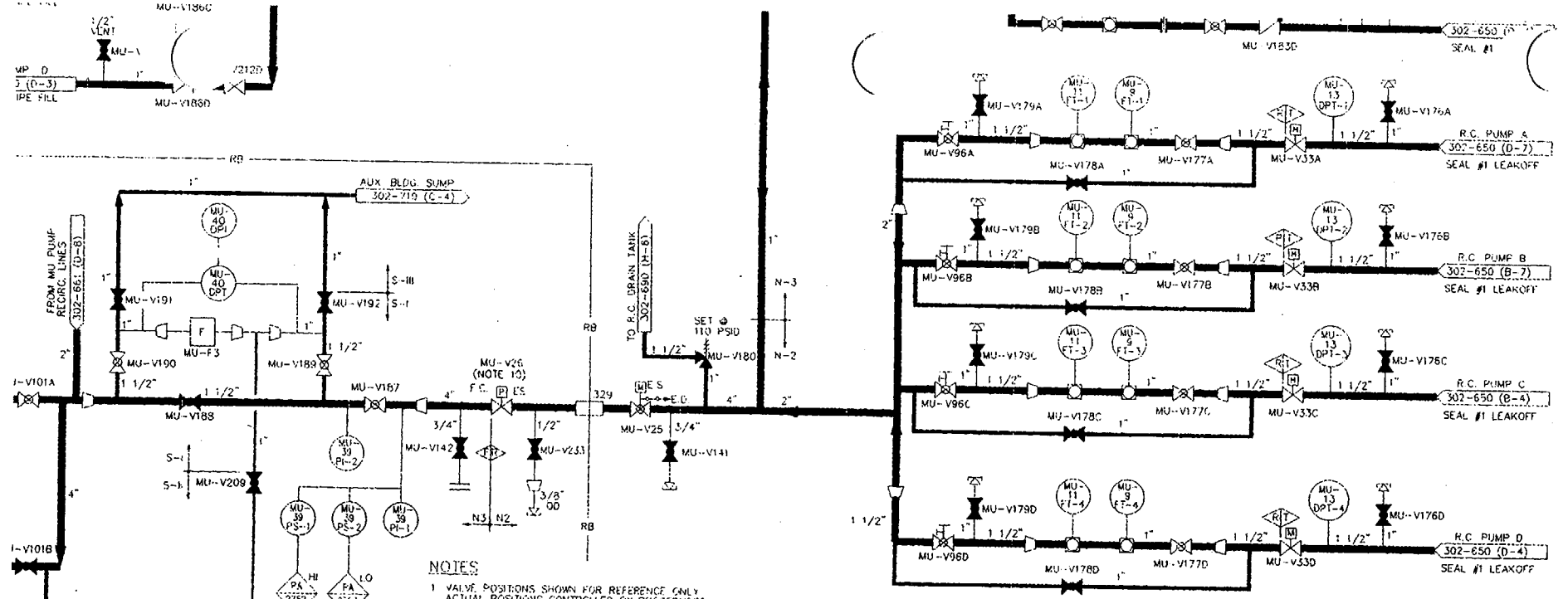
A Plausible because this instrument would swap the PORV to the low pressure setpoint if in auto.

B Plausible because other vents go to the RCDT, but these go to the RB atmosphere.

C Plausible because this instrument provides the low pressure input to the PORV.

D Correct answer. Upstream relief valve goes to RCDT.

Comments None.



NOTES

1. VALVE POSITIONS SHOWN FOR REFERENCE ONLY. ACTUAL POSITIONS CONTROLLED BY PROCEDURES
2. ALL INSTRUMENT TAG NUMBERS ARE PREFIXED BY SYSTEM DESIGNATOR UNLESS OTHERWISE NOTED
3. SET POINT VALUES FOR RELIEF VALVES ARE NOMINAL AND FOR REFERENCE ONLY. ACTUAL DATA [illegible]
4. [illegible]
5. VALVES MU-V[illegible] [illegible]
6. SYSTEM CLEANING REQUIREMENTS (FOR ERECTION ONLY)
   PIPING & COMPONENTS - CLASS C
   INSTRUMENTS & INSTRUMENT LINES - CLASS A
   OPEN END VENTS & DRAINS COMMERCIALLY CLEAN
   (X) RESIN ADD - SEE DWG 302-670
   (Y) RECLAIMED WATER - SEE DWG 302-670
   (Z) NUCLEAR SERVICES - SEE DWG 302-610
8. VENT CONNECTIONS ARE CAPPED WHEN VENT RIGS NOT IN PLACE.
9. SEE DWG. 302279 SHS 1&11 FOR INSTRUMENT AIR DETAILS TO VALVE OPERATOR
10. SEE DWG. 302279 SH.2 FOR INSTRUMENT AIR DETAILS TO VALVE OPERATOR
11. SEE DWG. 302279 SH.3 FOR INSTRUMENT AIR DETAILS TO VALVE OPERATOR
12. SEE DWG. 302279 SH.4 FOR INSTRUMENT AIR DETAILS TO VALVE OPERATOR
13. SEE DWG. 302279 SH.9 FOR INSTRUMENT AIR DETAILS TO VALVE OPERATOR
14. SEE DWG. 302279 SH.10 FOR INSTRUMENT AIR DETAILS TO VALVE OPERATOR
15. SEE DWG. 302279 SH.1 FOR INSTRUMENT AIR DETAILS TO VALVE OPERATOR

NO.	DWG. NO	TITLE
3	302-003	COMPONENT & SYS ID INDEX
2	302-002	SYMBOLS INSTRUMENTATION
1	302-001	SYMBOLS FLOW DIAGRAM

REFERENCES

THIS IS A COMPUTER GENERATED DRAWING DO NOT REVISE IT MANUALLY

GPU NUCLEAR / GAI

MAKEUP & PURIFICATION
FLOW DIAGRAM

TMI-1 | DWG. NO 302-660 | SH | REV. 38 | SCALE NONE

INTERFACING CONCURRENCE | GA NO

302-660

5 4 3 2 1

Form ES-401-6

Written Exam Question Worksheet

Q # 082

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 027 K5.01 Page 3.5-14 Tier # 2

RO/SRO Importance Rating 3.1 3.4 Group # 3

Measurement Knowledge of the operational implications of the following concepts as they apply to the CIRS: Purpose of charcoal filters

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

Which ONE (1) of the following are the charcoal filters in the RB Kidney Filter System designed to remove?

A. Iodine and Xenon

B. Xenon and Krypton

C. Iodine and particulates

D. Xenon and particulates

Technical Reference OP 1104-55, Reactor Building Atmosphere Clean-Up System, Rev. 21 page 6

Open Exam Reference

Learning Objective IV.F.02.02 State the functions of the Reactor Building Ventilation System.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .7

Discriminant Validity Statements

A Plausible, these are isotopes produced.

B Plausible, these are isotopes produced.

C Correct answer.

D Plausible, these are isotopes produced.

Comments None.

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GPU NUCLEAR	TMI Operating Procedure	Number 1104-55

..e

Reactor Building Atmosphere Clean-Up System		Revision No. 21
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: 09/07/99

This document is within QA plan scope		Yes	X	No
Safety Reviews Required		Yes	X	No

TRAINING DEPT.

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Originator		8/23/99
Procedure Owner		8-26-99
PRG		8/31/99
rover		8/27/99

3.3 Infrequent Operation

3.3.1 Reactor Building Atmosphere Clean-up

When the Reactor Building atmosphere is to be cleaned up, as determined by H.P. Analysis, and at least 4 hours prior to initiating R.B. Purge whenever iodine or particulates are >MDA.

____ 1. The Reactor Building Atmosphere Clean-up System is in Normal System Operation (Standby) per Section 3.2.

____ 2. Start Fan AH-E-101.

____ 3. Verify Low Flow Alarm is clear.

____ 4. When next Reactor Building entry is made, complete Enclosure 2, Section II, "Fan and Filter".

> **NOTE**
>
> Allow clean-up system to operate until R.B. air activity is within acceptable limits or as determined by Shift Supervisor.

____ 5. Notify Radiological Controls prior to securing AH-E-101.

> **NOTE**
>
> During a plant outage RB iodine activity can increase significantly when AH-E-101 is secured or trips resulting in evacuation of outage critical path workforce from the RB.

____ 6. When Reactor Building Atmosphere Clean-up System is no longer required, STOP AH-E-101.

3.3.2 Quarterly Surveillance

(Indicate satisfactory completion of steps below by initialing each step and sign name at end of appropriate section.)

I. Pumps

____ 1. Verify storage tank water level is visible in the top half of sight glass (if required, fill per Section 3.1.2).

____ 2. Verify a current system valve lineup is on file and that no outstanding tagging exists which will effect system operation.

3. Line up FS-P-5A/B for recirc operation:

____ a. Open FS-V-288

Form ES-401-6

Written Exam Question Worksheet

Q # 083

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 034 A3.02 Page 3.8-13 Tier # 2

RO/SRO Importance Rating 2.5 3.1 Group # 3

Measurement Ability to monitor automatic operation of the Fuel Handling System including: Load limits

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

Which ONE (1) of the following conditions would result in interlock actuation to prevent further insertion of a fuel assembly into a core location?

A. Grapple Tube Down indication.

B. Underload condition on the load cell.

C. Exceeding a lowering rate of 4 feet per minute.

D. The bottom of the fuel assembly reaches 3 feet above the core.

Technical Reference OPM section M-01 page 11 Rev. 6

Open Exam Reference

Learning Objective IV.B.15.19 Explain fuel hoist control, including:
c. What stops the fuel hoist when it is being lowered.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .5

Discriminant Validity Statements

A Plausible since this provides position indication, but no interlock action occurs.

B Correct answer.

C Plausible since there is an administrative lowering limit of 2 fpm when lowering the fuel assembly near other objects.

D Plausible since this is where the slow zone interlock comes into play using the Fuel Grapple Jog Control vice the Rod Hoist control. However, with fuel grappled, only the Fuel Grapple Jog Control is available for raising or lowering the cell.

Comments None.

FOR INFORMATION ONLY

SECTION M-01
REVISION 6

TMI OPERATIONS PLANT MANUAL
FUEL HANDLING SYSTEM

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Concur

Owner — Date 7/15/99

Operations Training Manager, TMI — Date 7/26/99

Training Coordinator, TMI — Date 8-3-99

Approved

Plant Operations Director TMI — Date 8/3/99

5.2.2 Interlocks

NOTE
No interlocks may be bypassed without express permission from the Fuel Handling Supervisor.

a. Hoist Down

In order for the hoist down circuit to be operable (for either the Fuel Hoist Control handle or the Fuel Grapple Jog Control) there must be: 1) An indication of the grapple being either engaged or disengaged, and 2) Air pressure 80 - 100 psig.

When the grapple is engaged an underload cutout is in effect to prevent setting the Fuel Assembly down in an improper manner or having the assembly hang up on an obstruction. This interlock will stop the hoist if the weight on the hoist falls below its underload cutout setpoint. The interlock is automatically bypassed by geared limit switches when the Fuel Assembly is at the bottom of the Storage Rack or Transfer Basket.

Both the grapple indication and the underload protection are negated when the hoist up and down interlock is bypassed. When the Fuel Mast is in the vicinity of a Transfer Trolley Basket, the hoist is inoperative unless that basket is in a vertical position. Hoist down operation is terminated by a low load cut-out when the weight on the hoist is reduced to approximately 600 lbs., which is approximately 200 lbs. less than the weight of the grapple alone. This protection cannot be bypassed.

b. Hoist Up

The hoist up circuit will not be operative on either the Fuel Hoist Control handle or the Fuel Grapple Jog unless there is an indication of grapple engaged or disengaged. This protection may be bypassed by the hoist up and down interlock. Air pressure should be 80 - 100 psig.

An overload cut-out provides protection against a Fuel Assembly becoming hung-up on an obstruction while it is being raised. This interlock will terminate hoist operation when the weight on the hoist reaches <u>about</u> 2600 lbs., which is <u>about</u> 180# over the weight of the grapple plus a Fuel Assembly plus a Control Rod Assembly, and there is a backup at 3000 lbs. This protection may be bypassed by the overload bypass switch. Hoist up operation will normally be terminated by the Grapple up disengaged if empty and up limit if loaded. When the grapple is disengaged, this limit will be satisfied when the grapple is withdrawn to a position approximately 2 feet into the mast where the Fuel Grapple unloaded slow zone is cleared. At this point the Fuel Hoist Control may be used to raise the grapple to the up limit.

During shipping cask loading, there may be a requirement to raise the grapple higher than normal to clear the top of the cask. In this case the fuel hoist interlock may be bypassed by operating TS-2. This will allow hoist up operation until a final limit switch is closed. At this point, the grapple cannot be raised any higher. Note though that if the protection basis trolley/bridge interlocks (TS-9/TS-3) are disabled also, then no interlock protection exist. Upward movement won't stop until impacting the mechanical stops on the hoist, the operator must stop the hoist manually in this situation.

Form ES-401-6

Written Exam Question Worksheet

Q # 084

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 041 A2.03 Page 3.4-25 Tier # 2

RO/SRO Importance Rating 2.8 3.1 Group # 3

Measurement Ability to predict the impacts of the following malfunctions or operations on the SDS; and based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Loss of IAS

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** C

Turbine Bypass Valve (TBV) H/A stations are in hand with demand set at approximately 20% open. A complete loss of the normal Secondary side Instrument Air header occurs.

Which ONE (1) of the following describes the response of the TBVs and the method to control the valves?

The valves will:

A. Open. Controlled locally using the handwheel.

B. Close. Controlled locally using the handwheel.

C. Remain 20% open. Controlled from the H/A station in the CR.

D. Go to 50% open. Controlled by adjusting local air pressure to the controller.

Technical Reference EP 1202-36, Loss of Instrument Air, page 9, Rev. 29.

Open Exam Reference

Learning Objective IV.C.01.04 Explain the operation of the turbine bypass valves (MS-V-3 A-F), including controls, setpoints and relief capacity.

Question Source ☑ New ☐ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge

☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .5 ☑ 55.43 .5 ☑ 55.45 .3, .13

Discriminant Validity Statements

A Plausible if fail position is misunderstood.

B Plausible if fail position is misunderstood.

C Correct answer. Backup air system will continue to supply motive air to the controller.

D Plausible if fail position is misunderstood.

Comments None.

FOR INFORMATION ONLY

GPU NUCLEAR	TMI Emergency Procedure	Number Q-084 1202-36
Title: **Loss of Instrument Air**		Revision No. **29**
Applicability/Scope: TMI Division	Responsible Office: Plant Ops. Director	Effective Date: **06/22/99**

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	David B. Wise	6/15/99
Procedure Owner	James N Paules	6/15/99
PRG	John S. LaRoche	6/15/99
Approver	James N Paules	6/15/99

TABLE 1

Air Operated Valve Number	Local Pressure at which component position becomes unreliable	Air Fail Position
IC-V-4	52#	Closed (Note 1)
IC-V-5	8#	Fail As Is
IC-V-6	46#	Closed (Note 1)
IC-V-74	46#	Open
MS-V-13A	24#	Open
MS-V-13B	24#	Open
MS-V-3A	27#	*Closed
MS-V-3B	27#	*Closed
MS-V-3C	27#	*Closed
MS-V-3D	27#	*Closed
MS-V-3E	27#	*Closed
MS-V-3F	27#	*Closed
MS-V-4A	27#	Closed**
MS-V-4B	27#	Closed**
MS-V-6	30#	Open**
MU-V-10	45#	Closed
MU-V-11A	38#	Closed
MU-V-11B	38#	Closed
MU-V-17	27#	Closed

* Supplied by backup air compressor. Will not go to failed position unless backup air is lost.

**Supplied by two hour back-up bottled air.

Form ES-401-6

Written Exam Question Worksheet

Q # 085

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 045 K4.42 Page 3.4-29 Tier # 2

RO/SRO Importance Rating 2.8 3.0 Group # 3

Measurement Knowledge of MT/G system design features and/or interlocks provide for the following: Operation of turbine bypass in event of load loss or plant trip

Proposed Question ☑ RO ☐ SRO ☑ PRA Related **Correct Answer** A

The plant has experienced a load rejection.

The following plant conditions exist:
- Turbine load is 50 Mwe
- The Diamond is in manual
- Reactor power has been stabilized at 20%

Which ONE (1) of the following design features will compensate for the mismatch between Reactor power and Turbine load?

A. The Turbine Bypass Valves will control Turbine Header pressure at 895 psig (setpoint + 10 psig bias).

B. The Turbine Bypass Valves will control Turbine Header pressure at 960 psig (setpoint + 75 psig bias).

C. The Atmospheric Dump Valves will control OTSG pressure between 1026 psig and 1052 psig.

D. The Main Steam Safety Valves will continue to lift to maintain OTSG pressure below 1040 psig.

Technical Reference AbP 1203-1, Load Rejection, Rev. 28 pages 2 and 4
OPM Section G-03, Main Steam, Rev. 13 pages 9 and 11

Open Exam Reference

Learning Objective IV.C.01.04 Explain the operation of the turbine bypass valves (MS-V-3 A-F), including controls, setpoints and relief capacity.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☐ 55.45

Discriminant Validity Statements

A Correct answer.

B Plausible because this would be the correct setpoint at 20% power if Reactor power and Turbine load was matched.

C Plausible if the examinee does determines that the mismatch is outside the capacity of the Turbine Bypass Valves.

D Plausible if the examinee does determines that the mismatch is outside the capacity of the Turbine Bypass and Atmospheric Dump Valves.

Comments None.

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AmerGen	TMI - Unit 1 Abnormal Procedure	Number 1203-1
Title **Load Rejection**		Revision No 29
Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date 02/22/00

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

SIMULATOR ROOM
FILE COPY

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Originator	Randy S. Campbell	2-8-00
Procedure Owner	Randy S. Campbell	2-14-00
PRG	[illegible]	2-16-2000
Approver	James [illegible]	2/15/2000

1.0 SYMPTOMS

1. Outgoing 230 KV line breakers open (indication on SS-1 panel).
2. Reactor power decreasing.
3. Increase in pressurizer level and RCS pressure.
4. Megawatts electric decreasing to auxiliary load. Reactor/Turbine Generator remain on line supplying 50 to 100 MWe.
5. High turbine header pressure, lifting of secondary steam safety valves.
6. Increase in generator frequency of 3 to 4 cycles.

2.0 IMMEDIATE ACTION

A. Automatic Action

1. The ICS trips to tracking because the turbine control goes to Local Control on power-load Unbalance Actuation.
2. ICS ULD runs back in tracking mode to 20 percent Reactor Power. (In approximately 4 minutes.)
3. EHC controls turbine speed to prevent overspeed trip of turbine.
4. Secondary safety valves and atmospheric dump valves open.
5. Turbine bypass valves open.
6. Pressurizer spray valve opens at 2205 psig.
7. Possible high pressure reactor trip at 2355 psig.

B. Manual Actions

> **NOTE**
>
> An asterisk (*) indicates that the parameter value must be reverified as the first step in the follow-up action. Use redundant indication where possible.

1. Operate pressurizer spray valve (RC-V-1) in manual if necessary to control RCS pressure.

*2. Verify feedwater flow and reactor power are being reduced or manually reduce to supply station service MWe load ~ 50.

_____ 3. When FW control reaches Lo Level Limits (H-1-6 and H-1-7) take manual control of the Rx Demand and maintain power at a level specified by the Control Room Supervisor.

4. Monitor pressurizer level, pressurizer temperature, RCS pressure, RCS temperature, OTSG level, and steam header pressure, and turbine speed.

_____ 4.1 Verify pressurizer heaters and spray control RC pressure at 2155 psig, if not, manually control.

> **NOTE**
>
> Pressurizer spray may have to be secured and pressurizer heaters manually turned on above 2155 psig RCS pressure if the pressurizer was subcooled due to the pressurizer insurge. This may prevent a low RCS pressure trip.

_____ 4.2 Adjust make up and letdown flows to control pressurizer level at 220 inches.

_____ 4.3 Verify steam header pressure is between 885 and 915 psig.

_____ 4.4 Verify feed flow is controlling OTSG level at ≥ 25 inches, if not, manually control.

_____ 4.5 Adjust turbine load set if required to maintain turbine speed at 1800 RPM.

_____ 5. Due to the buildup of Xenon to peak conditions a combination of rod withdrawal and deboration should be used to overcome peak Xenon.

> **NOTE**
>
> The diesel to be started should be dependent upon the current makeup pump lineup. If the makeup pump that is running is not ES selected, the makeup pump will TRIP when the bus feeder breaker is opened.

> **CAUTION**
>
> The time that the Diesel Generator is run unloaded should be minimized.

_____ 6. Start one diesel generator per Section 3.4 of 1107-3 and leave running continuously until two 230 KV lines are restored. (Ref. T.S. 3.7.2.e).

FOR INFORMATION ONLY

SECTION G-03
REVISION 13

TMI OPERATIONS PLANT MANUAL
MAIN STEAM SYSTEM

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Concur

Owner _____ Date 11/15/99

Operations Training Manager, TMI _____ Date 11/18/99

Training Coordinator, TMI _____ Date 11-30-99

Approved

Plant Operations Director TMI _____ Date 12/1/99

5.0 SYSTEM DESCRIPTION

The main steam system serves primarily to deliver steam from the steam generators to the high-pressure turbine of the turbine generator. It also provides steam during startup, shutdown, and normal operation to the main feedpump turbines, and during blackout or failure of both main feedpumps, to the emergency feedpump turbine which exhausts to atmosphere. There are two main steam leads from each steam generator to the turbine or a total of four lines. The only cross connection between leads is in the turbine steam chest between the turbine stop valves and control valves.

Since a portion of the main steam lines and the steam lines to the main feedpump turbines are located in the turbine hall which is not protected against hypothetical tornado, missile or aircraft incidents, main steam isolation stop check valves are located in the hardened portion of the Intermediate Building. These stop check valves are remotely closed by the operator from the control room, close in less than two minutes, and are tight closing.

The piping arrangement is such that rupture of a line from one steam generator will not blow the other steam generator dry, and ensure that a steam supply is available to the emergency feedpump turbine. Upstream of the main steam isolation valves are main steam safety valves and a steam supply to a divided header which serves:

- Steam bypass dump to the condenser (11.25% flow each side - total 22.5%). (3 valves per side) (MS-V-3A-F)
- Controlled steam relief to the atmosphere (3.2% flow each side - total 6.4%). (MS-V-4A, B)
- Steam supply to emergency feedpump turbine (required flow available from either side).

5.1 Modes of Operation - Normal

During normal operating conditions the main steam system supplies steam to the main turbine via the main turbine stop and control valves. A more detailed description of the main turbine along with the stop and control valves is contained in Section H-1 of this manual.

Also steam is supplied up to the main feed pump turbines through normally open valves MS-V-5A/B. The main feed pumps will be running on low pressure steam with the high pressure steam poppet valve closed. This arrangement provides high pressure steam to the feed pump turbines without delay should the low pressure steam become unavailable.

Steam is also supplied to the gland seal steam system via normally open valves MS-V-7 and GS-V-4. If pressure drops in the gland seal steam system a pressure regulating valve will open to return pressure to normal. Gland seal steam can also be supplied from auxiliary steam, and during high load operation from high pressure turbine seal leakoff.

During normal operation the steam generators are cross connected in only 2 places, the turbine valve chest and the relief valve post support heating system.

The relief valve post support heating system is supplied steam from each steam line via MS-V-85 A, B, C and D. The discharge of the main steam safety valves have supports which sit under the discharge line. When the steam system heats up the expansion of the piping causes the distance between the support and the relief discharge piping to increase. If this condition were left unchecked and the relief valve lifted the force exerted by the steam escaping through the discharge piping could create enough motion to crack the discharge piping.

The main steam safety valve setpoints are:

Valve No.	Size	Lift Press.	Rated Flow Per Valve at 3% Press. Accumulation
MS-V-21A, B	3 x 6 inch	1040 psig	194,900 lb/hr
MS-V-20B, C	6 x 10 inch	1092.5 psig	824,265 lb/hr
MS-V-19A, B, C, D	6 x 10 inch	1080 psig	814,955 lb/hr
MS-V-18A, B, C, D	6 x 10 inch	1060 psig	799,990 lb/hr
MS-V-17A, B, C, D	6 x 10 inch	1050 psig	792,610 lb/hr
MS-V-20A, D	6 x 10 inch	1050 psig	792,610 lb/hr

6.3 Atmospheric Dump Valves MS-V-4A/B

Each OTSG is provided with one atmospheric dump valve (MS-V-4A/B). This is a 6" pneumatically operated control valve that is rated for 3.2% steam flow. The atmospheric dump valves operate on a proportional basis. This means the valve opens proportional to the amount of overpressure, they start to open at 1026 psig and are fully open at 1052 psig. When the turbine bypass valves are bypassed, the atmospheric dump valves will open to maintain the OTSG pressure at 1010 psig.

These valves also can receive the automatic control signals normally sent to the turbine bypass valves (MS-V-3A-F). If main condenser vacuum drops to less than 23 inches Hg OR less than 2 circ water pumps are running, the turbine bypass valves are not permitted to operate and the control signals are passed to the MS-V-4A/B. These valves receive the same setpoint control signals as do MS-V-3A-F. These setpoints are described later.

6.4 Turbine Bypass Valves (MS-V-3A-F)

Each OTSG is provided with 3 Turbine Bypass Valves. The A OTSG has MS-V-3D, E, F while the B OTSG has MS-V-3A, B, C. The turbine bypass valves are 6 inch pneumatically operated control valves that are rated for 3.75% steam flow each.

These valves receive a control signal from their respective OTSG pressure signal. The control pressures are:

895 psig - To prevent bypass valves and turbine control valves from fighting against each other at low load operation. This setpoint is in effect when the Turbine is not tripped and unit load demand is less than 15% or if the reactor is not tripped and the Turbine is tripped.

960 psig - To prevent inadvertent bypass valve operation during normal plant transients. This setpoint is in effect if all Turbine bypass valves are closed and Turbine header pressure is within 10 psig of setpoint, or if Turbine header pressure is 10 psig greater than setpoint with unit load demand greater than 15%.

1010 psig - To prevent excessive pressurizer level decrease on reactor trip and cooldown. This pressure corresponds to a saturation temperature of 548°F. This setpoint is in effect when the reactor is tripped.

1026 psig - High pressure relief, open proportionally, start to open at 1026 psig fully open 1052 psig even if in hand in control room.

Form ES-401-6

Written Exam Question Worksheet

Q # 086

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 078 K3.02 Page 3.8-19 Tier # 2

RO/SRO Importance Rating 3.4 3.6 Group # 3

Measurement Knowledge of the effect that a loss or malfunction of the IAS will have on the following: Systems having pneumatic valves and controls

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** C

RB emergency cooling was placed in operation in response to a steam leak inside containment. The air supply line to RR-V-6 fails resulting in a loss of all air to RR-V-6.

Which ONE (1) of the following actions will be required to maintain proper flow and header pressure in the Reactor River Water system?

A. Open RR-V-6.

B. Throttle RR-V-5.

C. Close RR-V-6 and throttle RR-V-5.

D. Open RR-V-6 and throttle RR-V-5.

Technical Reference EP 1202-36, Loss of Instrument Air, page 7 rev 29.

Open Exam Reference

Learning Objective IV.D.14.09 Determine the operability of plant systems and explain the required course of action, if any, given:
a. the status of the Instrument Air system and components in systems supplied with Instrument Air;
b. the plant condition (e.g., power operations);
c. appropriate references (e.g. Tech Specs, procedures and/or flow diagrams/prints).

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .7 ☐ 55.43 ☑ 55.45 .6

Discriminant Validity Statements

A Plausible if failure mode of RR-V-6 is unknown. RR-V-6 will fail full open on a loss of air.

B Plausible if failure mode of RR-V-6 is unknown.

C Correct answer.

D Plausible if failure mode of RR-V-6 is unknown.

Comments None.

FOR INFORMATION ONLY

GPU NUCLEAR	TMI Emergency Procedure	Number 1202-36
Title: **Loss of Instrument Air**		Revision No. 29
Applicability/Scope: TMI Division	Responsible Office: Plant Ops. Director	Effective Date: 06/22/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	David B. Wilson	6/15/99
Procedure Owner	James N Paules	6/15/99
PRG	John S. Lahoda	6/15/99
Approver	James N Paules	6/15/99

CAUTION

After loss of 2-hour Backup Air System with local accumulator air available, RR-V-6 fails open. Establish communication with Control Room, locally close RR-V-6 and use RR-V-5 to maintain cooler outlet pressure 55-60 psig.

After loss of 2-hour Backup Air System and loss of local accumulator air, RR-V-6 fails open. Establish communication with Control Room, locally close RR-V-6 and use RR-V-5 to maintain cooler outlet pressure 55-60 psig.

If for any reason RR-V-6 fails closed, locally close RR-V-6 and use RR-V-5 to maintain cooler outlet pressure 55-60 psig.

_____ 24. If ES has actuated or RB Temp. approaches 130°F, establish RB Emergency Cooling and verify cooler outlet pressure 55-60 psig.

_____ 25. Review Table 1 and verify control air is available and each component is in the desired position and functioning properly.

NOTE

It is not desirable to cooldown without IA available. This should only be accomplished if IA can not be restored in a reasonable time frame. Consideration should be given to available air sources, time to cooldown and ability to maintain hot shutdown.

_____ 26. If instrument air can not be restored, cooldown using emergency feedwater and natural circulation per OP 1102-11 "Plant Cooldown", Enclosure 2.

Form ES-401-6

Written Exam Question Worksheet

Q # 087

Examination Outline Cross-Reference RO SRO

KA # SYSTEM 103 K1.08 Page 3.5-18 Tier # 2

RO/SRO Importance Rating 3.6 3.8 Group # 3

Measurement Knowledge of the physical connections and/or cause-effect relationships between the containment system and the following systems: SIS, including action of safety injection reset

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** A

The plant was operating at 100% power when the following sequence of events occurred:
- SBLOCA
- RB pressure rose to 5 psig
- RCS pressure dropped to 400 psig
- All systems responded in automatic as required
- Manual initiation of ESAS was NOT performed

Following completion of the IMAs for the appropriate ATOG procedures, the following plant conditions are observed:
- RB pressure stabilized at 1 psig
- RCS pressure is 950 psig and rising slowly at 20 psig every 5 minutes

In order to regain control of ESAS actuated components, IAW OP 1105-3, Safeguards Actuation System, for each train the operator must push:

A. ALL 3 4 psig Enable Reset push buttons.
ALL 3 LPI Enable & Channel Reset push buttons
ALL 3 HPI Bypass push buttons

B. ALL 3 4 psig Enable Reset push buttons.
ALL 3 LPI Bypass push buttons
ALL 3 HPI Bypass push buttons

C. 2 of 3 4 psig Defeat push buttons
ALL 3 LPI Enable & Channel Reset push buttons
ALL 3 HPI Bypass push buttons

D. 2 of 3 4 psig Defeat push buttons
ALL 3 LPI Bypass push buttons
ALL 3 HPI Bypass push buttons

Technical Reference OP 1105-3, Safeguards Actuation System, pages 14, 15, and 16, Rev 41.

Open Exam Reference

Learning Objective IV.E.24.16 Using console indication, demonstrate or describe how to bypass, reset or defeat an ESAS auto actuation. This is to include HPI, LPI, RB isolation and cooling, RB high high pressure and reactor trip isolation.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .2 TO .9 ☐ 55.43 ☑ 55.45 .7, .8

Discriminant Validity Statements

A Correct answer.

B Plausible if examinee fails to recognize that pressure is above the bypass enable setpoint.

C Plausible if examinee fails to recognize the procedure requirements to depress all 3 4 psig Enable Reset

push buttons.

D Plausible if examinee fails to recognize that pressure is above the bypass enable setpoint and the procedure requirements to depress all 3 4 psig Enable Reset push buttons.

Comments None.

CONTROLLED COPY

AmerGen	TMI - Unit 1 Operating Procedure	Number 1105-3
Title **Safeguards Actuation System**		Revision No. 41
Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date 02/29/00

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

TRAINING DEPT.

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	Signature	Date
Originator	[illegible]	2/21/00
Procedure Owner	Thomas H. Hoalley	2/21/00
PRG	John S. Schah	2/22/2000
Approver	James N. Paulles	2/22/2000

b. With the reactor reset, press the 3 Green Enable buttons on CR for RT1B, RT2B, RT3B and the 3 Green Enable Buttons on CC for RT1A, RT2A, RT3A. Verify the blue lights on PCR labeled RT1A, 2A, 3A and RT1B, 2B, 3B are out. Individual valves may now be operated with their normal controls.

2. If Required By Plant Emergency. or Operating Procedures and with the Shift Manager's/Control Room Supervisor's permission the 4 PSI R.B. Isolation System may be defeated/reset/enabled as follows:

a. Manual Actuation

> NOTE
>
> The Manual DEFEAT/ENABLE pushbuttons alternate between the "DEFEAT" position and the "ENABLE" position each time they are depressed.

1. Defeat the 4 psi Manual Actuation by going to "DEFEAT" with all 3 of the DEFEAT/ENABLE buttons on console CC labeled PB2/RBA, PB3/RBA, PB4/RBA and on console CR labeled PB2/RBB, PB3/RBB, PB4/RBB.

2. Verify the yellow "DEFEAT" light on each of the pushbuttons is lit.

3. Verify the 4 psi Manual Actuation lights (6 total) on panel PCR are out.

4. If not automatic actuation signal is present, then the individual components may now be operated with their normal controls.

> NOTE
>
> All 3 'DEFEAT" lights for the respective train must be lit prior to returning that train to "ENABLE".

5. After all of the pushbuttons have been placed into DEFEAT, the Manual Actuation System should be enabled, unless plant conditions do not permit, by going to "ENABLE" with all 3 of the DEFEAT/ENABLE buttons on console CC labeled PB2/RBA, PB3,RBA, PB4/RBA and on console CR labeled PB2/RBB, PB3/RBB, PB4/RBB.

6. Verify that the green "ENABLE" light on each of the pushbuttons is lit.

b. Automatic Actuation

1. With RB Pressure > 4 PSI press 2 of the 3 Amber Defeat Buttons on CR for RB1B, RB2B, RB3B and 2 of the 3 Amber Defeat Buttons on CC for RB1A, RB2A, RB3A. Verify the corresponding blue lights on PCR for RB1A, 2A, 3A and RB1B, 2B, 3B (3 lights per channel) are out. The individual valves may now be operated with their normal controls.

2. With RB Pressure < 4 PSI, reset all 3 of the 4 PSI logic channels by pressing the 3 green enable buttons on CR for RB1B, RB2B, RB3B and by pressing the 3 green enable buttons on CC for RB1A, RB2A, RB3A. Verify the blue lights on PCR for RB1A, 2A, 3A and RB1B, 2B, 3B are all out. The individual valves may now be operated with their normal controls.

3. If Required by Plant Emergency or Operating Procedures and with the Shift Manager's/Control Room Supervisor's permission, H.P. Injection may be Bypassed/Reset as follows:

> NOTE
>
> If 4 psi R.B. Isolation Auto or Manual Actuation has occurred, the 4 psi actuation(s) must be reset/defeated/enabled prior to resetting/defeating/enabling LPI and/or HPI Actuation Signal(s).

a. Manual Actuation

> NOTE
>
> The Manual DEFEAT/ENABLE pushbuttons alternate between the "DEFEAT" position and the "ENABLE" position each time they are depressed.

1. Defeat the 1600 psi Manual Actuation by going to "DEFEAT" with all 3 of the DEFEAT/ENABLE buttons on console CC labeled PB2/RCA, PB3/RCA, PB4/RCA and on console CR labeled PB2/RCB, PB3/RCB, PB4/RCB.

2. Verify the yellow "DEFEAT" light on each of the pushbuttons is lit.

3. Verify the 1600 psi Manual Actuation Lights (6 total) on panel PCR are out.

4. If no automatic actuation signal is present, then the individual components may now be operated with their normal controls. Refer to Attachment 2 for additional guidance for the restoration of components.

NOTE

All 3 "DEFEAT" lights for the respective train must be lit prior to returning that train to "ENABLE".

5. After all of the pushbuttons have been placed into DEFEAT, the Manual Actuation System should be enabled, unless plant conditions do not permit, by going to "ENABLE" with all 3 of the DEFEAT/ENABLE buttons on console CC labeled PB2/RCA, PB3/RCA, PB4/RCA and on console CR labeled PB2/RCB, PB3/RCB, PB4/RCB.

6. Verify that the green "ENABLE" light on each of the pushbuttons is lit.

b. Automatic Actuation

1. With RCS Pressure < 1725 PSI and decreasing depress the three HP. Inj. Bypass Pushbuttons on CR, and the three "HP Inj. Bypass" Pushbuttons on CC. Verify the blue lights on PCR labeled RC1A, 2A, 3A and RC1B, 2B, 3B are out. Refer to Attachment 2 for guidelines for restoring HPI components.

2. With RCS Pressure >1670 PSI and steady or increasing check the HPI actuation bistables reset and reset the HP Injection actuation relays by pressing the 3 green Enable and Channel reset buttons for RC1B, RC2B, RC3B on CR and the 3 similar buttons on CC for RC1A, RC2A, RC3A. Verify the blue lights on PCR labeled RC1A, 2A, 3A and RC1B, 2B, 3B are out. Refer to Attachment 2 for guidelines for restoring HPI components.

4. If required by Plant Emergency. or Operating Procedures and with the Shift Manager's/Control Room Supervisor's permission L.P. Injection may be bypassed/reset as follows:

a. Manual Actuation - refer to 3.5.3 for resetting HPI (1600 psi manual only)

b. Automatic Actuation

1. With RCS pressure < 875 PSI and decreasing press the 3 Amber Bypass Buttons on CR for RC4B, RC5B, RC6B and the 3 Amber Bypass Buttons on CC for RC4A, RC5A, RC6A. Verify the blue lights on PCR labeled RC4A, 5A, 6A and RC4B, 5B, 6B are out. Refer to Attachment 2 for guidance for restoring HPI components.

2. With RCS Pressure > 570 PSI and steady or increasing press the 3 Green Enable and Channel Reset Buttons on CR for RC4B, RC5B, RC6B and the 3 Green Enable and Channel Reset Buttons on CC for RC4A, RC5A, RC6A. Verify the blue lights on PCR labeled RC4A, 5A, 6A and RC4B, 5B, 6B are out. Refer to Attachment 2 for guidelines for restoring components.

Form ES-401-6

Written Exam Question Worksheet

Q # 088

Examination Outline Cross-Reference RO SRO

KA # Conduct of Operations 2.1.3 Page 2-1 Tier # 3

RO/SRO Importance Rating 3.0 3.4 Group #

Measurement Knowledge of shift turnover practices.

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

Which ONE (1) document is NOT required to be reviewed prior to assuming shift duties as a licensed Control Room Operator?

A. ESAS Checklist.

B. Locked Valve Book.

C. TCN/STP Book

D. Revision Review Book.

Technical Reference AP 1012, Shift Relief and Log Entries, Rev 45, page 9.

Open Exam Reference

Learning Objective V.A.06.04 State the documents which will be reviewed prior to and after assuming the shift per AP 1012, Shift Relief and Log Entries.

Question Source ☐ New ☑ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☑ Used in Last Two TMI NRC Exams Exam Date 1998 SRO EXAM

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .10 ☐ 55.43 ☑ 55.45 .13

Discriminant Validity Statements

A Plausible misconception – this document is required to be reviewed prior to assuming the duty.

B Correct answer.

C Plausible misconception – this document is required to be reviewed prior to assuming the duty.

D Plausible misconception – this document is required to be reviewed prior to assuming the duty.

Comments None.

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AmerGen	TMI - Unit 1 Administrative Procedure	Number 1012
Title **Shift Relief and Log Entries**		Revision No. 45
Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date 03/03/00

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	D Jones	2-18-00
Originator	[illegible]	2/22/00
Procedure Owner	W J Marshall for J R Powers	2/24/00
PRG	John J. Schab	2/25/2000
Approver	[illegible]	2/28/00
Approver	John S Cotton	3/1/2000

4.5.2.4 Start and stop of any planned or unplanned, radioactive or significant non-radioactive. liquid or gas release. Log release permit number or release point, type and quantity.

4.5.2.5 When problems or abnormalities are encountered, a narrative of the event is to include relevant information and sufficient detail to reconstruct the event.

4.5.3 The Group Operations Supervisor shall periodically review and sign the Primary Auxiliary Operator Log, verifying an appropriate and correct level of detail describing shift activities.

4.6 Shift Relief

4.6.1 All shift operations personnel shall be responsible for maintaining their duty station until properly relieved. The Shift Manager, Control Room Supervisor, Control Room Operators and Auxiliary Operators shall be relieved by qualified personnel only e.g., those personnel who are properly licensed and properly informed of the plant status, operations in progress, technical specification equipment out of service and any special instructions which may be applicable.

4.6.2 Prior to relieving the shift the relieving individual will discuss plant status, operations in progress, shift turnover checklist.

4.6.3 The oncoming CRO, CRS and SM will each sign the ES Checklist to indicate his understanding.

4.6.4 Prior to assuming the shift, the relieving individual shall review station logs, records, special instructions, etc., which have been generated since his last shift. The logs and records to be reviewed should include:

4.6.4.1 Control Room Log (Senior Reactor Operator and Control Room Operator).

4.6.4.2 TCN and STP Books (Senior Reactor Operator and Control Room Operator).

4.6.4.3 Operations Memo Book (Senior Reactor Operator and Control Room Operator).

4.6.4.4 Revision Review Book (Senior Reactor Operator and Control Room Operator).

4.6.4.5 TMI Night Orders (Senior Reactor Operator and Control Room Operator).

4.6.5 The following documents shall be reviewed during the shift.

4.6.5.1 Active Tagging Application Book (SRO and CRO)

4.6.5.2 Locked Valve Log (SRO and CRO)

4.6.5.3 Outstanding Surveillance Schedule (SRO)

Form ES-401-6

Written Exam Question Worksheet

Q # 089

Examination Outline Cross-Reference RO SRO

KA # Conduct of Operations 2.1.25 Page 2-3 Tier # 3

RO/SRO Importance Rating 2.8 3.1 Group # 1

Measurement Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data.

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

Plant conditions:
- Maximum Generation Emergency declared.
- Reactor power is 100%.
- Generator MVAR loading is 460 MVARS OUT.
- Main Generator Hydrogen pressure is 60 psig.

Refer to the attached Figure B-2B from OP 1106-1. From the list of values provided below, identify the MAXIMUM GENERATOR MEGAWATT OUTPUT CAPABILITY for these conditions.

A. 0 MW.

B. 600 MW.

C. 800 MW.

D. 870 MW.

Technical Reference OP 1106-1, Turbine Generator, Rev. 94, Figure B-2B (Rev. 94), Page 59.

Open Exam Reference OP 1106-1, Turbine Generator, Rev. 94, Figure B-2B (Rev. 94), Page 59.

Learning Objective IV.C.09.20 Explain the relationship between main generator H2 pressure and allowable generator electrical load.

Question Source ☐ New ☑ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .10 ☑ 55.43 .5 ☑ 55.45 .12

Discriminant Validity Statements

A This is the correct answer for -460 MVARS.

B This is the correct answer using the wrong (45 psig) curve.

C Correct answer.

D This is rated load for the unit.

Comments Parent question was not used during program quizzes nor Audit Exams.

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AmerGen

TMI - Unit 1
Operating Procedure

Number: **1106-1**

Title: **Turbine Generator**

Revision No: **95**

Applicability/Scope: TMI Division

Responsible Office: Director, Operations

Effective Date: 03/29/00

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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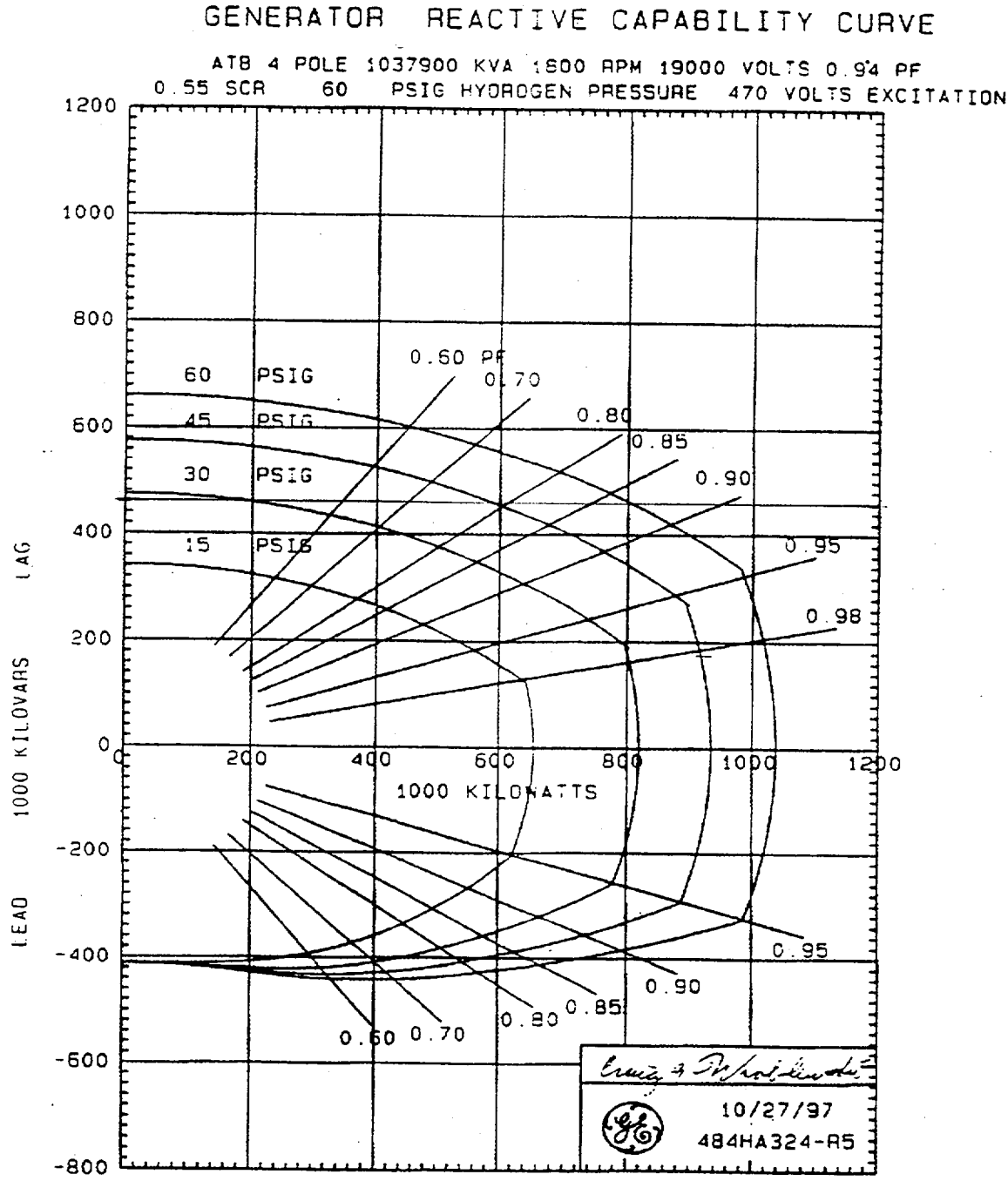
	Signature	Date
Originator	George J Bilar	03/14/2000
Procedure Owner	Charles C. Seitz	3/16/00
PRG	Howard R Crawford PRG Vice Chairman	3/22/00
Approver	[illegible]	3/19/00

FIGURE B-2B


GENERATOR REACTIVE CAPABILITY CURVE
ATB 4 POLE 1037900 KVA 1800 RPM 19000 VOLTS 0.94 PF
0.55 SCR 60 PSIG HYDROGEN PRESSURE 470 VOLTS EXCITATION
1200
1000
800
600
400
200
0
-200
-400
-600
-800
1000 KILOVARS
LAG
LEAD
0
200
400
600
800
1000
1200
1000 KILOWATTS
60 PSIG
45 PSIG
30 PSIG
15 PSIG
0.60 PF
0.70
0.80
0.85
0.90
0.95
0.98
0.95
0.90
0.85
0.80
0.70
0.60
10/27/97
484HA324-R5

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AmerGen	TMI - Unit 1 Operating Procedure	Number 1106-1
Title Turbine Generator		Revision No. 95
Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date 03/29/00

This document is within QA plan scope [X] Yes [ ] No
Safety Reviews Required [X] Yes [ ] No

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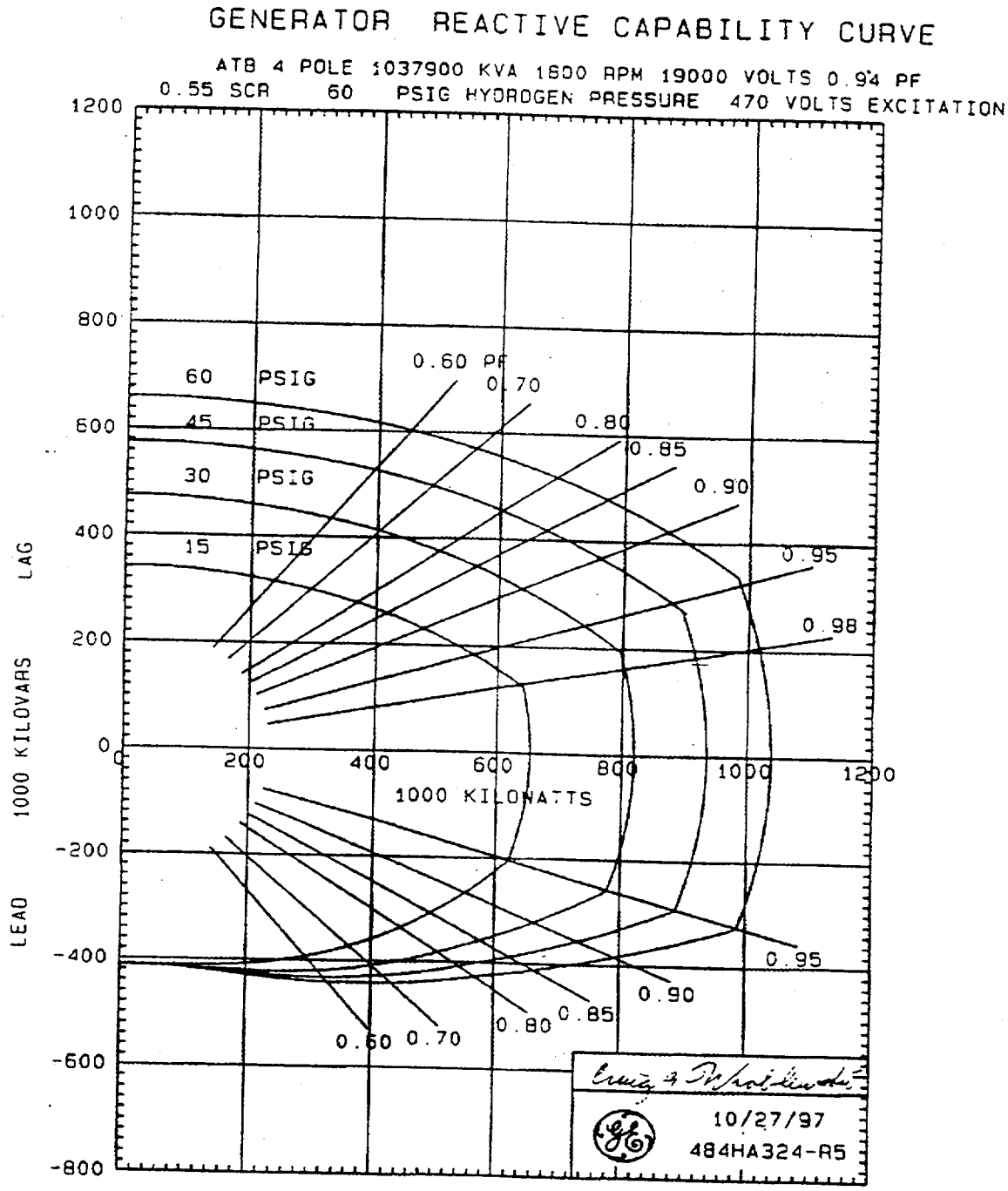
	Signature	Date
Originator	George J Bilar	03/14/2000
Procedure Owner	Charles C. Seitz	3/16/00
PRG	Howard Crawford PRG Vice Chairman	3/22/00
Approver	[illegible]	3/19/00

FIGURE B-2B


GENERATOR REACTIVE CAPABILITY CURVE
ATB 4 POLE 1037900 KVA 1800 RPM 19000 VOLTS 0.94 PF
0.55 SCR 60 PSIG HYDROGEN PRESSURE 470 VOLTS EXCITATION
1200
1000
800
600
400
200
0
-200
-400
-600
-800
LAG
1000 KILOVARS
LEAD
200
400
600
800
1000
1200
1000 KILOWATTS
60 PSIG
45 PSIG
30 PSIG
15 PSIG
0.60 PF
0.70
0.80
0.85
0.90
0.95
0.98
0.95
0.90
0.85
0.80
0.70
0.60
10/27/97
484HA324-R5

Form ES-401-6

Written Exam Question Worksheet

Q # 090

Examination Outline Cross-Reference RO SRO

KA # Conduct of Operations 2.1.29 Page 2-4 Tier # 3

RO/SRO Importance Rating 3.4 3.3 Group # 1

Measurement Knowledge of how to conduct and verify valve lineups: performance of valve lineup - independent verification of valve position.

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

From the list below, identify the ONE (1) statement that describes the requirements for performing an Independent Verification for an easily accessible manual valve that is required to be closed.

A. One individual closes the valve, second individual verifies from an independent remote position DEMAND indicator that the valve closed.

B. One individual closes the valve, second individual independently uses a remote POSITION indicator to verify the valve is closed.

C. One individual closes the valve, second individual independently verifies the valve is closed by physically turning in the closed direction.

D. One individual closes the valve, while the second individual visually observes the first individual closing the valve.

Technical Reference AP 1067, Independent Verification Program, Rev. 29, Page 7.

Open Exam Reference

Learning Objective V.J.01.02 State how valve position must be verified for valves without remote position indication/ and state when valve position may be verified by remote indication - according to AP 1067 Independent Verification.

Question Source ☐ New ☑ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question # 88

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .10 ☐ 55.43 ☑ 55.45 .1, .12

Discriminant Validity Statements

A Plausible misconception that use of DEMAND indication is an acceptable means of position verification.

B Plausible misconception that use of remote position indication is acceptable for manual valves with easy access.

C Correct answer.

D Plausible distracter since this is procedurally acceptable for valves which are not easily accessed.

Comments 1998 TMI SRO Exam Question 33.

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AmerGen	TMI - Unit 1 Administrative Procedure	Number 1067
Title **Independent Verification Program**		Revision No. 29
Applicability/Scope TMI Division	Responsible Office Plant Operations Dir.	Effective Date 03/03/00

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

TRAINING DEPT.

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	Signature	Date
Originator		2/9/00
Procedure Owner	James N. Paulls	2/22/2000
PRG	Jack S. Fahah	2/23/2000
Approver		2/28/00
Approver	John B. Cotton	3/1/2000

4.3 Independent Verification of Components

Components shall be independently verified to be in their proper alignment by the following guidelines:

a. Switches, breakers must be independently checked by two (2) separate personnel visually sighting these components in the proper position. Remote light indicators are acceptable where supplied for verification use. If this verification is being performed after the completion of a Surveillance Test, the person doing the independent verification must be different from the person who signed the first position check as part of the conduct of the surveillance procedure. The independent verifier should have the task requirement or procedure available to ensure the action or task was conducted on the correct equipment, train or string.

b. Verification of valve positions will be as follows:

> CAUTION
>
> Remote indication shall only be used when indication is direct component position as opposed to power demand indication. In all cases the ES status panel indication is acceptable.

1. If the valve has remote indication, two (2) independent personnel may use that remote indication for their verification.

2. For manual valves, two (2) independent checks of position must be performed. Each party will verify its proper position by physically turning the valve in the closed direction. If a valve is found to be in an unexpected position, inform the Control Room Supervisor or Shift Manager and obtain further guidance before changing the valve position. For valves that are difficult to get to (i.e., need ladders to reach or other physical difficulties) it is permissible for one person to operate the valve and a second person to perform the second verification visually. This option will only be accepted as the second check when the second party can by his proximity certify operation was sufficient to determine proper valve position, and the second person has independently verified, with the procedure or task requirement, that the valve being checked is the correct valve.

5.0 RESPONSIBILITIES

5.1 V.P. and Director TMI-1 - Responsible for the overall direction and implementation of the independent verification program.

5.2 Plant Manager - Responsible to ensure department managers are familiar with independent verification practices and requirements.

5.3 Plant Maintenance Director - Responsible to ensure program is properly administered on the departmental level including establishing training requirements for those personnel involved.

5.4 Manager Plant Training - Ensure appropriate personnel are adequately trained in accordance with requirements provided by the Plant Maintenance Director and Plant Operations Director.

Form ES-401-6

Written Exam Question Worksheet

Q # 091

Examination Outline Cross-Reference RO SRO

KA # Conduct of Operations 2.1.32 Page 2-4 Tier # 3

RO/SRO Importance Rating 3.4 3.8 Group #

Measurement Ability to explain and apply all system limits and precautions.

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

The Borated Water Storage Tank has the following conditions present:
- Boron concentration is 2450 ppm
- Water temperature is 45 degrees F

Which ONE (1) of the following describes the required action and the basis for that action?

A. Restore boron concentration to at least 2500 ppm in order to ensure that adequate boron concentration exists when filling the fuel transfer canal for refueling.

B. Restore boron concentration to at least 2500 ppm in order to ensure that reactor remains subcritical by at least 1% DK/K following a LOCA.

C. Restore water temperature to at least 50 degrees F to prevent crystallization of the boron in the delivery system.

D. Restore water temperature to at least 50 degrees F to prevent thermal shock to the HPI nozzles.

Technical Reference TS 3.3.1.1 and bases.
OP 1103-4, Soluble Poison Control, page 6, Rev 50.

Open Exam Reference

Learning Objective IV.B.22.01 State the Technical Specifications for:
b. BWST, Core Flood, and Spent Fuel Boron concentrations

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .10 ☑ 55.43 .2 ☑ 55.45 .12

Discriminant Validity Statements

A Plausible since boron concentration during refueling has minimum limits.

B Correct answer.

C Plausible if lower temperature limit is misinterpreted.

D Plausible if lower temperature limit and reason for limit is misinterpreted.

Comments None.

FILE COPY

GPU NUCLEAR	TMI Operating Procedure	Number Q-91 1103-4

Title

Soluble Poison Concentration Control

Revision No. **50**

Applicability/Scope	Responsible Office	Effective Date
TMI Division	Plant Operations Dir.	10/16/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

TRAINING DEPT.

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Originator	R. Mopasso for PSB	10/16/99
Procedure Owner	James N Carles for BLATR	10/16/99
PRG	[illegible]	10/16/99
Approver	James N Carles	10/16/99

2.0 LIMITS AND PRECAUTIONS

2.1 Equipment

1. When running boric acid injection pumps CA-P-1A, CA-P-1B, insure a flow path is set up through the normal injection route of WDL-V-61 and batch totalizer controller MU12-FT or through WDL-V-62 to the R.C. Bleed Tank or via the emergency injection route through MU-V-51.

2. Refer to Operating Procedures Below for Applicable Equipment Limits and Precautions.

   OP 1104-29E Bleed and Feed Processes, OP 1104-47B Chemical Addition Nuclear.

   OP 1104-2 Makeup and Purification System.

   OP 1103-2 Filling and Venting the Reactor Coolant System.

2.2 Administrative

1. The reactor shall not be critical unless the Borated Water Storage Tank contains a minimum of 350,000 gallons of water having a minimum concentration of 2,500 ppm boron at a temperature not less than 40°F.

2. The BAMT or RBAT's may be used for emergency boration if the following are met:

   a. The boric acid mix tank contents should have the equivalent boron per the COLR at a temperature of at least 10°F above the crystallization temperature with an operable system (including one pump) to deliver the solution to the makeup system. The temperature requirement for the system piping and valves should be at least the same temperature as that required for the mix tank.

   b. A reclaimed boric acid storage tank should have the equivalent of Boron per the COLR, subject to the same conditions as the boric acid mix tank.

3. The CRA safety groups will always be at their upper limits prior to boron dilution for approach to criticality, per T.S. 3.1.3.5.

4. Terminate boration or dilution if neutron count rate, control rod indications, or other reactivity indications behave in an erratic or unexpected manner.

5. When performing a significant change in boron concentration, sample letdown line every hour. While subcritical, insure sampling is performed approximately every 30 ppm boron.

6. If actual boron concentration deviates from expected by more than 50 ppm boron, determine the cause.

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS

Applicability

Applies to the operating status of the emergency core cooling, reactor building emergency cooling, and reactor building spray systems.

Objective

To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building emergency cooling and reactor building spray systems.

Specification

3.3.1 The reactor shall not be made critical unless the following conditions are met:

3.3.1.1 Injection Systems

a. The borated water storage tank shall contain a minimum of 350,000 gallons of water having a minimum concentration of 2,500 ppm boron at a temperature not less than 40°F. If the boron concentration or water temperature is not within limits, restore the BWST to OPERABLE within 8 hrs. If the BWST volume is not within limits, restore the BWST to OPERABLE within one hour. Specification 3.0.1 applies.

b. Two makeup pumps are operable in the engineered safeguards mode powered from independent essential buses. Specification 3.0.1 applies.

c. Two decay heat removal pumps are operable. Specification 3.0.1 applies.

d. Two decay heat removal coolers and their cooling water supplies are operable. (See Specification 3.3.1.4) Specification 3.0.1 applies.

e. Two BWST level instrument channels are operable.

f. The two reactor building sump isolation valves (DH-V6A/B) shall be remote-manually operable. Specification 3.0.1 applies.

3.3.1.2 Core Flooding System

a. Two core flooding tanks each containing 940 ± 30 ft^3 of borated water at 600 ± 25 psig shall be available. Specification 3.0.1 applies.

b. Core flooding tank boron concentration shall not be less than 2,270 ppm boron.

c. The electrically operated discharge valves from the core flood tank will be assured open by administrative control and position indication lamps on the engineered safeguards status panel. Respective breakers for these valves shall be open and conspicuously marked. A one hour time clock is provided to open the valve and remove power to the valve. Specification 3.0.1 applies.

3.3.3 Exceptions to 3.3.2 shall be as follows:

a. Both core flood tanks shall be operable at all times.

b. Both the motor operated valves associated with the core flood tanks shall be fully open at all times.

c. One reactor building cooling fan and associated cooling unit shall be permitted to be out-of-service for seven days.

3.3.4 Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be verified to be operable.

Bases

The requirements of Specification 3.3.1 assure that, before the reactor can be made critical, adequate engineered safety features are operable. Two engineered safeguards makeup pumps, two decay heat removal pumps and two decay heat removal coolers (along with their respective cooling water systems components) are specified. However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both core flooding tanks are required because a single core flooding tank has insufficient inventory to reflood the core for hot and cold line breaks (Reference 1).

The operability of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA (Reference 2). The limits on BWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident (LOCA).

The contained water volume limit of 350,000 gallons includes an allowance for water not usable because of tank discharge location and sump recirculation switchover setpoint. The limits on contained water volume, NaOH concentration and boron concentration ensure a pH value of between 8.0 and 11.0 of the solution sprayed within containment after a design basis accident. The minimum pH of 8.0 assures that iodine will remain in solution while the maximum pH of 11.0 minimizes the potential for caustic damage to mechanical systems and components. Redundant heaters maintain the borated water supply at a temperature greater than 40°F.

Form ES-401-6

Written Exam Question Worksheet

Q # 092

Examination Outline Cross-Reference RO SRO

KA # Equipment Control 2.2.11 Page 2-6 Tier # 3

RO/SRO Importance Rating 2.5 3.4 Group # 2

Measurement Knowledge of the process for controlling temporary changes.

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

To support plant start-up, the valve line-up of a system is required to be modified until work is completed on that system. From the list below, identify the ONE (1) correct description of approval(s) that satisfies requirements to permit this valve line up change, in accordance with AP 1001A.

A. Two Licensed CROs.

B. Shift Manager with an SRO License.

C. One Licensed CRO AND the on-shift STA.

D. One Licensed CRO AND one Shift Manager/Control Room Supervisor with an SRO license.

Technical Reference AP 1001A, Procedure Review and Approval, Rev. 41, Page 19.

Open Exam Reference

Learning Objective V.A.02.04 Explain the requirements for making temporary changes to valve line-ups per AP 1001A.

Question Source ☐ New ☑ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .10 ☑ 55.43 .3 ☑ 55.45 .13

Discriminant Validity Statements

A Plausible since both CROs are licensed.

B Plausible since the Shift Manager is the senior shift management representative responsible for all operations.

C Plausible misconception since this incorrect answer involves two individuals concurring independently.

D Correct answer.

Comments

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AmerGen

TMI - Unit 1
Administrative Procedure

Number: **1001A**

Title: **Procedure Review and Approval**

Revision No.: **41**

Applicability/Scope: TMI Division

Responsible Office: Vice President TMI - Unit 1

Effective Date: **01/11/00**

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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Originator	[illegible]	1-10-2000
Procedure Owner	[illegible]	1-10-2000
PRG	[illegible]	1-10-2000
Approver	[illegible]	1/10/2000
Approver	John B Cotton	1/10/2000

4.5.11 Maintenance Supervision shall ensure previous and current revisions of a Job Ticket Written Procedure are kept with the work package.

4.5.12 Maintenance Supervision shall ensure final disposition of completed Job Order Written Procedures in accordance with appropriate record keeping and filing procedures.

4.6 Preparation, Review and Approval of Temporary Change Notices (TCN's)

4.6.1 Use of the TCN - Use of the TCN shall be limited to the following cases:

4.6.1.1 To ensure orderly and uniform operations for short periods when the unit, a system, or a component of a system is performing in a manner not covered by existing (standing) detailed procedures or has been modified or extended in such a manner that portions of existing procedures do not apply.

4.6.1.2 When the change is of a temporary nature and should not result in a permanent revision.

4.6.1.3 When time or plant conditions do not permit the use of a Procedure Change Request.

> NOTE
>
> Temporary Changes to the valve lineups required by Procedure Valve Checklists **do not** require the use of a TCN. The deviations must be noted in indelible marker on the applicable check off sheet and must be reviewed and approved by two Licensed Reactor Operators, one of whom must be a Shift Manager/Control Room Supervisor with an SRO License. The deviations must be noted in the Control Room Log.

4.6.2 Types of TCN's

> NOTE
>
> Change in intent may be indicated by any of the following:
>
> - Proposed change goes beyond the purpose or scope of the current procedure.
> - The Safety Determination for the proposed change requires a detailed Safety Evaluation to support the proposed change.
> - A change in the method of performing the procedure if the potential adverse consequences could be significant.

4.6.2.1 "Normal" Approval Route

This approval route should be used for the majority of TCN's. This route is identical to the review and approval route of PCR's in that all reviews and approvals are obtained prior to implementation. The "normal" approval route is usually used when a change is required to be implemented on short notice.

Form ES-401-6

Written Exam Question Worksheet

Q # 093

Examination Outline Cross-Reference RO SRO

KA # Equipment Control 2.2.12 Page 2-6 Tier # 3

RO/SRO Importance Rating 3.0 3.4 Group # 2

Measurement Knowledge of surveillance procedures.

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

Plant conditions:
- Reactor power is 100%.

During performance of SP 1301-1, Shift and Daily Checks, RM-A-4P failed to meet the acceptance criteria.

Identify the ONE (1) statement that describes PROCEDURALLY REQUIRED reactor operator actions.

A. Apply a CAUTION TAG to RM-A-4P.

B. Apply a DO NOT OPERATE TAG to RM-A-4P.

C. Complete a SDR (Surveillance Deficiency Report) sheet.

D. Notify the Radcon Department that RM-A-4P is inoperable.

Technical Reference SP 1301-1, Shift and Daily Checks, Rev. 136, Pages 10 and 20.

Open Exam Reference

Learning Objective V.A.04.15 Given a situation or condition which could arise during Tech. Spec. surveillance testing, properly document any deficiencies on a Surveillance Deficiency Report (SDR) per AP-1001J, Technical Specification Surveillance Testing Program.

Question Source ☐ New ☑ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .10 ☑ 55.43 .3 ☑ 55.45 .13

Discriminant Validity Statements

A Plausible misconception regarding application of Caution Tags.

B Plausible misconception regarding application of DNO Tags.

C Correct answer.

D Plausible misconception regarding sampling requirements.

Comments Stem slightly modified to preclude eliminating a distractor in question 33.

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AmerGen	TMI - Unit 1 Surveillance Procedure	Number 1301-1
Title **Shift and Daily Checks**		Revision No. 136
Applicability/Scope TMI Division	Responsible Office Director, Operations	Effective Date 03/28/00

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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PRG	[illegible]	3/28/2000
Approver	[illegible]	3-28-00

D. Returning Digital Readout to μCi/cc

1) Place Program Mod Keyswitch to "Enter Data".

2) Depress "DISP OP" button.

3) Depress "0" button.

4) Depress "ENTER" button.

5) Place Program Mod Keyswitch to "RUN".

6) Remove key and return to locked key locker.

8.5 Perform Enclosure 4 each shift when RCS is less than 250°F. This is an operability verification to comply with Tech. Spec. 3.4.2.

9.0 ACCEPTANCE CRITERIA

9.1 All data evaluation should be answered "Yes". Complete a "SDR" sheet for all "No" answers.

9.2 All readings are within applicable COLR and/or Tech Spec limits.

10.0 POST MAINTENANCE TESTING

None

DATA SHEET 4

Daily Checks (Night Shift)

Sections A, C, D, E, G, H, I, and K to be completed at 2100 ± 2 hours

8.4A ODCM Part I, 3.1.2 and 3.1.1 **(to be completed at 2100 ± 2 hours).**

When required, perform source check and verify that recorder & meter indications are comparable.

Required	Channel	Record (CPM) Background	Expected Increase Above Bkgrd	Source Check OK? Circle Y or N	PPC/Recorder Pt. vs Meter OK? Check √ (OK) PPC Pt./Recorder	
At all times	RM-A4P		400	Y/N		
	RM-A4I		80	Y/N		
	RM-A4G		150	Y/N		
At all times	RM-A6P		200	Y/N		
	RM-A6I		100	Y/N		
	RM-A6G		200	Y/N		
At all times	RM-A8P		30	Y/N		
	RM-A8I		40	Y/N		
	RM-A8G		70	Y/N		
During RB purge (NIS if RB purge is not in progress)	RM-A9P		40	Y/N/NIS		
	RM-A9I		17	Y/N/NIS		
	RM-A9G		180	Y/N/NIS		
If condenser vacuum is established	RM-A5		147	Y/N		
	RM-A15		104	Y/N		
During ESF vent operation (NIS if EST vent is not operating)	RM-A14G		>800	Y/N/NIS		
	RM-A14G Hi		>4 80 x 10^{-9} amps	Y/N/NIS		
At all times	RM-A7					
	RM-L6		1450	Y/N		
Not T.S. required	RM-L7		1400	Y/N		
During CCB vent operation (NIS if CCB vent is not operating)	ALC-RMI-18P		200	Y/N/NIS		
	ALC-RMI-18I		200	Y/N/NIS		
	ALC-RMI-18G		200	Y/N/NIS		

NOTE

Attach printout of Area 21, Group 13 or Area 19, Groups 1, 3, 4, 5.

If required (see Sect. 8.4.3.2) was RM-L6 source check per Sect. 8.4.3.2.a acceptable? (Circle Y/N) Y/N/Not Required

Performed By_____ Time:_____

Form ES-401-6

Written Exam Question Worksheet

Q # 094

Examination Outline Cross-Reference RO SRO

KA # Equipment Control 2.2.13 Page 2-6 Tier # 3

RO/SRO Importance Rating 3.6 3.8 Group # 2

Measurement Knowledge of tagging and clearance procedures.

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

In addition to the person having clearance, identify ONE (1) specific person whose approval is required prior to changing the condition or position of BLUE tagged ES equipment (SINGLE manipulation).

A. Plant Manager.

B. Director, Operations.

C. Duty Shift Manager.

D. Licensed Control Room Operator.

Technical Reference AP 1002, Rules for Protection of Employees Working on Electrical and Mechanical Apparatus, Rev. 86, Page 9.

Open Exam Reference

Learning Objective V.A.01.02 State whose permission is required to change the position/condition of blue tagged equipment IAW AP 1002.

Question Source ☐ New ☑ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question #

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .10 ☐ 55.43 ☑ 55.45 .13

Discriminant Validity Statements

A A plausible misconception as this is a senior management official.

B A plausible misconception as this is a senior management official.

C Correct answer.

D Plausible misconception since this individual is licensed by the NRC.

Comments 1998 TMI SRO Exam Question 11. Shift Supervisor title was recently changed to Shift Manager, Shift Foreman titles were recently changed to Control Room Supervisor and Group Operations Supervisor - procedure updates are not complete. Answer reference was updated from Rev. 85 to Rev. 86 with no impact on answer - 2/8/00.

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AmerGen	TMI - Unit 1 Administrative Procedure	Number 1002
Title **Rules for the Protection of Employees Working on Electrical and Mechanical Apparatus**		Revision No. 87
Applicability/Scope TMI Division	Responsible Office Plant Manager	Effective Date 02/04/00

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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Procedure Owner	[illegible]	1/14/2000
PRG	[illegible]	1/20/2000
Approver	RE Frank	1-24-2000
Approver	John B Cotton	1-31-2000

4.3.3 BLUE tagged equipment CAN be energized or operated, but ONLY with permission of individual having clearance on BLUE tag, AND duty Shift Manager/Control Room Supervisor.

> **NOTE**
>
> Permission to energize or operate BLUE tagged equipment should be granted only when required to support work for which application was submitted. Otherwise, BLUE tags should be cleared and removed per Sections 4.15 to 4.17 of this procedure before changing equipment condition.

❶ Duty Shift Manager/Control Room Supervisor shall ensure record of each and every BLUE tagged equipment condition change is logged in Control Room Supervisor's logbook. The duty Shift Manager/Control Room Supervisor may specify a single logbook entry for multiple changes of condition of a BLUE tagged component within a time period specified for those changes.

4.3.4 Duty Shift Manager/Control Room Supervisor will specify condition equipment is to be left in after requested operations are completed. (Ex: MU-V-16A to be left closed after cycling).

❶ IF requested condition is different from specified on BLUE tag, duty Shift Manager/Control Room Supervisor should evaluate BLUE tag and remove per Section 4.15 through 4.17 of this procedure, if possible.

4.4 System Outages (Not SOTA)

> **NOTE**
>
> Component and component string outages performed by Plant Maintenance while operating and shutdown are called "System Outages." System outages are completed within a short time, typically two to three days.
>
> Scope of work is identified on a single system outage schedule which includes tasks to be performed by Electrical, I & C, Mechanical, Utility and other personnel that may be preventive, corrective, modifications, Technical Specifications, or special tests or checkouts.

4.4.1 One individual on each shift during which work is done shall be identified as Task Coordinator.

❶ Task Coordinator shall obtain clearance on application identifying himself as the Task Coordinator to the SWT Operator (or designated alternate).

❷ Task Coordinator shall ensure required protection provided by tagging application and clearance control document for ALL work on system outage schedule.

❸ IF required, Task Coordinator shall change tagging application to provide protection for all work on system outage schedule including task addition.

Form ES-401-6

Written Exam Question Worksheet

Q # 095

Examination Outline Cross-Reference RO SRO

KA # Radiation Control 2.3.1 Page 2-9 Tier # 3

RO/SRO Importance Rating 2.6 3 Group # 3

Measurement Knowledge of 10 CFR: 20 and related facility radiation control requirements.

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** C

The following conditions exist in a localized area in the Auxiliary Building:
- General area dose rate is 15 mrem per hour
- Highest contaminated swipe survey reads 40,000 dpm/100 square centimeters beta-gamma AND 2100 dpm/100 square centimeters alpha.
- Airborne activity will result in 20% of a DAC in one hour.
- The area is posted as a TLD Required Area.

Which ONE (1) of the following identifies the additional posting required for this area?

A. Contaminated Area; Airborne Radioactivity Area

B. Radiation Area; Airborne Radioactivity Area

C. Radiation Area; High Contamination Area

D. High Radiation Area; Contaminated Area

Technical Reference GET Employee Training, Radiation Worker Training, pages 56 and 59, Rev. 5.

Open Exam Reference

Learning Objective Define, identify the postings, and state the requirements to enter each of the following radiological areas:
High Contamination Area
Radiation Area

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge
☐ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .12 ☑ 55.43 .4, .9 ☑ 55.45 .9, .10

Discriminant Validity Statements

A Plausible since contamination and airborne activity exist.

B Plausible since radiation levels and airborne activity exist.

C Correct answer.

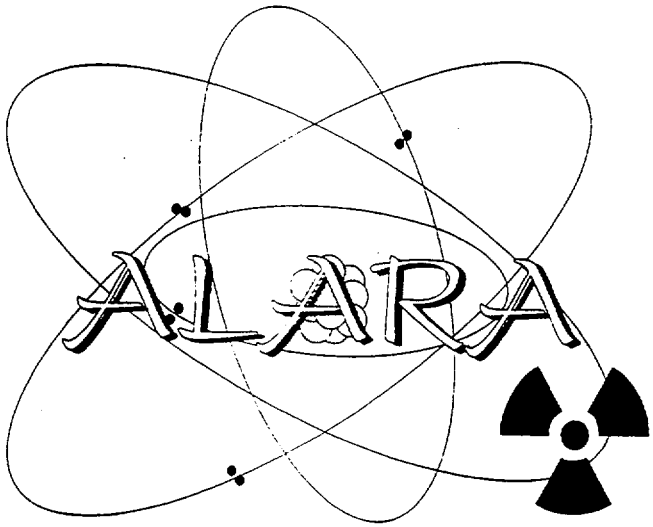
D Plausible since contamination areas and radiation levels exists.

Comments

Q-095

GPU NUCLEAR

General Employee Training

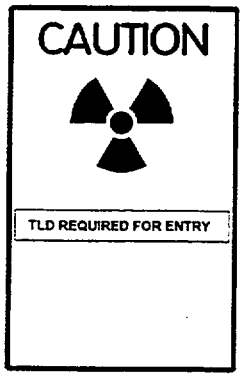


Radiation Worker Training

Radiologically Controlled Area (RCA) is an area for which access is limited for the purpose of protecting individuals from radiological hazards. There are many types of RCAs found at OC and TMI; each of these is posted to protect individuals from radiation exposure, radioactive contamination, or airborne radioactivity. **At OC**, the term "the RCA" is used to refer to the contiguous areas of the plant where radiological controls are in effect.

Most RCAs at OC and TMI are also Radiation Work Permit (RWP) areas, where an RWP is required for entry. Entry requirements for each RWP area are defined by the associated RWP. Personnel should read each caution sign prior to entering any area.

The following section lists the various types of RCAs found at OC and TMI. For each of these areas, the minimum identification requirement is shown. Where more than one type of radiological hazard is present, additional inserts may be added to the posting.

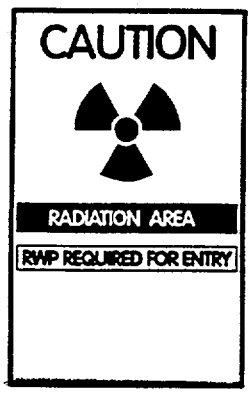


TLD Required Area:

This is an area with a dose rate greater than (or equal to) 0.5 mrem per hour. In general, an RWP is not required for entry.

Radioactive Materials Area:

This is an area that has radioactive materials stored in it. An area is required to be posted as a Radioactive Materials Area if it contains licensed material in an amount exceeding 10 times the quantity specified in 10CFR20 Appendix C.



Radiation Area:

This is an area with a dose rate greater than (or equal to) **5 mR per hour** at 30 cm (~1 ft) from the source.

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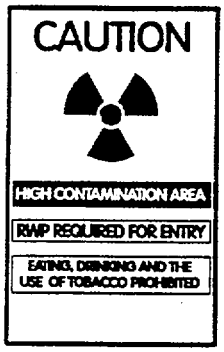
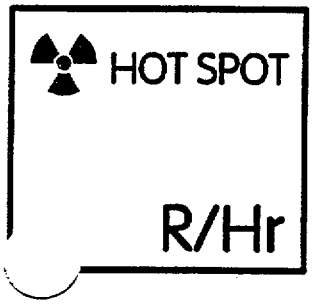


High Contamination Area

This is an area that has greater than 100,000 DPM/100 cm^2 of beta-gamma contamination or 2,000 DPM/100 cm^2 of alpha contamination. Entering this area requires authorization by RWP. The RWP will specify entry requirements.

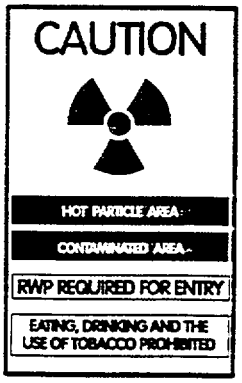
Other types of postings are used to warn of specific radiological hazards:



Hot Spot:

At TMI, a hot spot is defined as a localized area in which the deep dose contact reading is greater than (or equal to) 400 mR per hour (if located outside of posted high radiation areas.)

At OC, a hot spot is defined as a localized area in which the deep dose contact reading is greater than (or equal to) 100 mR per hour (if located outside of posted high radiation areas.) Hot spots should be avoided in order to maintain worker dose ALARA.



Hot Particle Area:

This is an area established to control the spread of hot particles.

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Form ES-401-6

Written Exam Question Worksheet

Q # 096

Examination Outline Cross-Reference RO SRO

KA # Radiation Control 2.3.2 Page 2-9 Tier # 3

RO/SRO Importance Rating 2.5 2.9 Group # 3

Measurement Knowledge of facility ALARA program.

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

A job is to be performed in an area that has a general radiation dose rate of 60 mRem/hr. Installation of temporary shielding to reduce the dose rate to 10 mRem/hr will require a total dose of 30 mRem (for installation and removal).

From the list below, identify the ONE (1) option that will meet ALARA expectations for the entire job process.

A. 2 workers each taking 35 minutes to perform the job.

B. 1 worker taking 60 minutes to perform the job.

C. Utilize temporary shielding, 2 workers each taking 35 minutes to perform the job.

D. Utilize temporary shielding, 1 worker takes 60 minutes to perform the job.

Technical Reference GET Radiation Worker Training Handout, Rev. 5, Page 19.

Open Exam Reference

Learning Objective III.F.01.08, Explain the ALARA program used at TMI-1.
III.F.01.09 Explain how time, distance and shielding can be used to minimize radiation exposure.

Question Source ☐ New ☑ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☑ Used in Last Two TMI NRC Exams Exam Date FEB 2000 CRO
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .12 ☑ 55.43 .4 ☑ 55.45 .9, .10

Discriminant Validity Statements

A 70 mRem total.

B 60 mRem total.

C 30 + 11.6 = 41.6 mRem total.

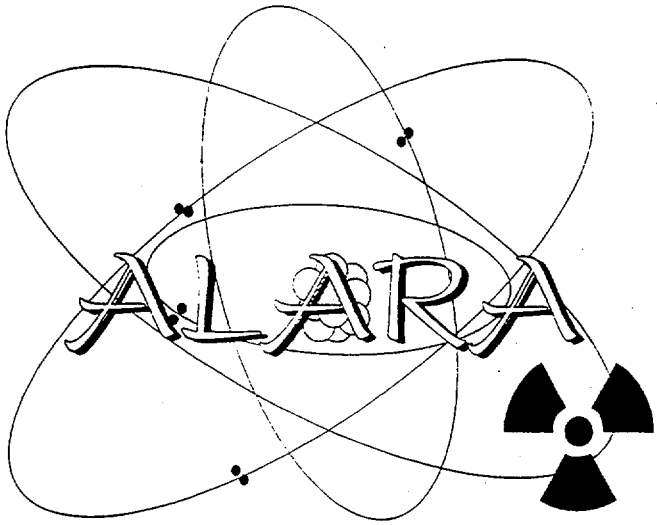
D Correct answer (30 + 10 = 40 mRem total).

Comments None.

Q-096

GPU NUCLEAR

General Employee Training



Radiation Worker Training

Distance from sources of radiation can be increased through several methods:

- using long-handled tools
- working at arm's length from the radiation source
- performing lengthy repairs in lower dose rate areas
- avoiding hot spots

LO: Explain how shielding may be used to reduce dose and state methods to implement this ALARA factor.

Just as using a potholder to pick up a scorching hot pot handle will decrease the amount of heat felt by your hand, increasing the amount of shielding between you and your co-workers and sources of radiation will decrease the dose you and your co-workers receive from the radiation sources.

For example: A valve emitting a large amount of radiation can be shielded to reduce the amount of dose received by a worker near the valve.

Examples of shielding include:

- Permanent shielding such as walls, steel support structures and permanently mounted equipment
- Temporary shielding such as lead blankets, lead bricks and concrete blocks
- At TMI: Point-of-work shielding such as lead blankets applied to certain plant components to reduce dose rates during a specific, short duration task.

LO: State the individual's responsibilities regarding temporary shielding.

The following precautions apply to the use of temporary shielding to reduce dose rates in an area:

- Temporary shielding will only be used when it provides for a net decrease in worker dose. The ALARA concept does not apply just to the individual, but to the plant as a whole. If shielding will reduce an individual's dose by 30 mrem, but will cause other workers to receive 50 mrem during the installation/removal of the shielding, there will be a net **INCREASE** of 20 mrem for the station.

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Form ES-401-6

Written Exam Question Worksheet

Q # 097

Examination Outline Cross-Reference RO SRO

KA # Radiation Control 2.3.4 Page 2-9 Tier # 3

RO/SRO Importance Rating 2.5 3.1 Group #

Measurement Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** B

Identify the ONE (1) statement that describes the MINIMUM requirement(s) for an individual to be allowed to receive a TEDE dose of 4100 mRem per year, excluding a planned special exposure.

A. A special RWP is written covering the individual to be permitted to exceed 4000 mRem.

B. Approvals from the Director, Rad Health & Safety AND the Vice President, TMI-1.

C. Approval from the President of the Company.

D. Notification of the NRC.

Technical Reference GET Radiation Worker Training Handout, Rev. 5, Page 14.

Open Exam Reference

Learning Objective III.F.02.01 State the Federal and TMI-1 Administrative Limits on Radiation Exposure at TMI-1 for whole body, skin and extremities.

Question Source ☐ New ☑ TMI Bank TMI Question #

☐ Modified TMI Bank Parent Question # 6

Question History ☑ Used in Last Two TMI NRC Exams Exam Date 1998 SRO EXAM

☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☑ Memory or Fundamental Knowledge

☐ Comprehension or Analysis

10 CFR Part 55 Content ☐ 55.41 ☑ 55.43 .4 ☑ 55.45 .10

Discriminant Validity Statements

A Plausible misconception that a special RWP would permit additional exposure.

B Correct answer.

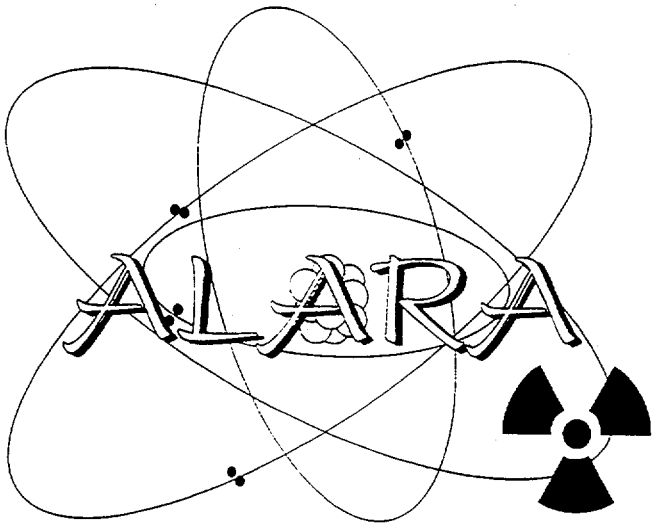
C Plausible misconception since this individual is required to pre-approve exposures in excess of 4500 mRem.

D Plausible misconception that the NRC could authorize additional exposure.

Comments 1998 TMI SRO Exam Question 6.
1/19/00 - Edited question stem to specify MINIMUM requirements and 4100 mRem exposure. This positively eliminated distracter C as a potentially correct answer. This individual COULD approve this exposure in the absence of the individuals in the normally correct answer (B). This individual is REQUIRED to approve exposures in excess of 4500 Mrem.

GPU NUCLEAR

General Employee Training



Radiation Worker Training

.ese GPU Nuclear administrative dose limits* are:

TEDE	4 rem/year	extremities	40 rem/year
lens of the eye	12 rem/year	any internal organ	40 rem/year
skin of the whole body	40 rem/year		

* Note that these limits are set to 80% of the federal limits.

LO: Discuss how company administrative dose limits are extended.

When a worker is approaching the TEDE dose limit, the Radiological Controls Management can extend the worker's dose limit. Extensions can be granted as long as the individual will NOT exceed federal limits. The following is a list of the possible extensions:

➢ Exceeding **4 rem/year** requires the approval of:

- **TMI:** the Radiological/Environmental Health & Safety Director and the Site Director
- **OC:** the Director, Radiological Health & Safety and the Vice President/Director, Oyster Creek

At TMI, exceeding **4.50 rem/year** requires the approval of the President, GPU Nuclear.

LO: Compare company dose administrative limits to federal dose limits.

GPU Nuclear has established administrative exposure/dose limits at 80% of the limits set by the federal regulations. This is to ensure that workers keep their doses As Low As Reasonably Achievable (ALARA) for the work assigned in radiation areas.

REMEMBER: Everyone has a part in ALARA!

LO: State the company emergency dose limits.

During an emergency, the Emergency Director can authorize an individual to exceed the legal dose limits. Recall that the Emergency Director is the individual in charge of plant activities during an emergency condition.

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Form ES-401-6

Written Exam Question Worksheet

Q # 098

Examination Outline Cross-Reference RO SRO

KA # Emerg Procedures/Plan 2.4.11 Page Tier # 3

RO/SRO Importance Rating 3.4 3.6 Group #

Measurement Knowledge of abnormal condition procedures.

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** D

The following plant conditions exist:
- Plant power is 90% and rising at 1% per minute.
- RCS pressure is 2100 psig and dropping 5 psig per minute
- Average RCS temperature is 578 degrees F and dropping at 1 degree per minute
- MU-V-17 is opening to maintain pressurizer level at 220 inches
- Generator load is 760 MWE and decreasing at 15 MWE per minute
- Alarm B-2-8, RB Clrs Excess Condensate, is in
- RB pressure is 0.5 psig and rising at 0.1 psig per minute

Which ONE (1) of the following is the required response to these conditions?

A. Trip the reactor and go to 1210-1.

B. Reduce power to <45% and trip the turbine.

C. Commence a one hour RCS leak rate calculation.

D. Commence plant shutdown at the rate specified by the SM.

Technical Reference AbP1203-24, Steam Leak, Rev. 27 pages 2 and 3.

Open Exam Reference

Learning Objective V.C.10.03 Given a list of symptoms, identify the event, determine what Immediate Automatic Actions will occur and what Immediate Manual Actions are required IAW AP 1203-24.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .10 ☑ 55.43 .5 ☑ 55.45 .13

Discriminant Validity Statements

A Plausible since this is an IMA choice determined by leak location.

B Plausible since this is an IMA choice determined by leak location.

C Plausible since this would address the RCS conditions.

D Correct answer.

Comments None.

FOR INFORMATION ONLY

GPU NUCLEAR	TMI Abnormal Procedure	Number 1203-24
Title: **Steam Leak**		Revision No. 27
Applicability/Scope: TMI Division	Responsible Office: Plant Ops. Director	Effective Date: 03/31/98

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

List of Effective Pages

Page	Revision	Page	Revision	Page	Revision	Page	Revision
1	27						
2	27						
3	27						
4	27						

	Signature	Date
Originator	James A. Stubbs	3/23/98
Procedure Owner	[illegible]	3-23-98
PRG	[illegible]	3-25-98
Approver	G. Noll	3-24-98

1.0 **SYMPTOMS**

1. Decreasing secondary steam pressure.
2. Electrical load reducing (mismatch between electrical load and Rx Power).
3. Decrease in pressurizer level, R.C. Pressure, and cold leg temperature.
4. For a leak inside the Reactor Building; Indication of increasing Reactor Building pressure and temperature.
5. For a leak outside the Reactor Building; Noise may be heard in Control Room or a report made from personnel outside the Control Room or M.S. Line Acoustic Monitor Alarm.

2.0 **IMMEDIATE ACTION**

A. Automatic Action

1. If HSPS MFW Isolation actuates (<600 psig) on the affected OTSG (could be both), the following valves auto close.
   a. Startup Feedwater Control Valve FW-V-16A(B)
   b. Main Feedwater Control Valve FW-V-17A(B)
   c. Main Feedwater Block Valve FW-V-5A(B)
   d. Startup Feedwater Block Valve FW-V-92A(B)
2. Possible Reactor trip on low pressure.

B. Immediate Manual Action

1. If the steam leak is upstream of the turbine stop valves (or the leak location is unknown) and either:
   a. HSPS actuates on either SG <u>or</u>
   b. Continued operation presents a hazard to personnel or equipment required for safe shutdown.

   Then manually trip the reactor and go to ATP 1210-1.
2. If the steam leak is downstream of the turbine stop valves and time permits.

   Then reduce power to < 45 percent and manually trip the turbine. Continue to reduce reactor power to hot standby.
3. If a. Continued operation is not posing a hazard to personnel or equipment required for safe shutdown but is severe enough to require shutdown <u>or</u>

b. RB pressure exceeds 2 psig.

Then reduce load at rate specified by SS/SF and go to following steps.

3.0 FOLLOW-UP ACTION

The objective of this procedure is to continue to shutdown and cooldown the unit while monitoring the steam leak.

_____ 1. Continue to shutdown and cooldown the unit per OP 1102-10 and 11 respectively.

_____ 2. Determine which OTSG has the steam leak and if possible the location of the leak. If possible, the leak should be isolated.

> **NOTE**
>
> If at anytime during the unit shutdown the steam leak degrades to where personnel or plant equipment required for safe shutdown are threatened then trip the reactor and go to ATP 1210-1.

_____ 3. Monitor RB Pressure and Temperature if leak is in the RB.

4. If the steam leak is in the RB, start containment cooling to keep RB Pressure < 2.0 PSIG and RB temperature < 130°F (if necessary) as follows:

_____ a. Start RR-P-1A (RR-P-1B) and insure RR-V-10A (RR-V-10B) open.

_____ b. Open RR-V-3A [and RR-V-3B if needed].

_____ c. Open RR-V-4A [and RR-V-4B if needed].

_____ d. Open RR-V-1A (RR-V-1B) and insure RR-V-10A (RR-V-10B) shuts.

_____ e. Close NS-V-85 and insure NS-V-8 is closed.

_____ f. Verify RR-V-6 is controlling backpressure at ≥ 55# or control system pressure by jog button for RR-V-5 (55-60 psig) as indicated on Console Right.

_____ g. If additional cooling is needed start the second RR pump and open RR-V-3C and RR-V-4C.

_____ h. Monitor RR cooler outlet flow (Computer Points A1049, A1050, A1051) and RB sump level to detect cooler leakage.

_____ 5. If steam leak is in RB, drain RB sump as needed to prevent overflowing the sump. (Containment Isolation Valve opening criteria must be met).

> **NOTE**
>
> The following step takes into consideration the assumption that a Main Steam Line Break inside of the Reactor Building has the potential to rupture an "RR" cooling coil.

Form ES-401-6

Written Exam Question Worksheet

Q # 099

Examination Outline Cross-Reference RO SRO

KA # Emerg Procedures/Plan 2.4-18 Page 2-13 Tier # 3

RO/SRO Importance Rating 2.7 3.6 Group #

Measurement Knowledge of the specific bases for EOPs.

Proposed Question ☑ RO ☑ SRO ☐ PRA Related **Correct Answer** A

The plant was operating at 100% power when the following sequence of events occurred:
- 3 gpm OTSG tube leak is detected.
- Prior to implementing the ATOG procedures, a major steam line rupture occurs in the RB.
- Current RB pressure is 5 psig and rising at 2 psig per minute.
- All systems operated as expected for this condition.

Which ONE (1) of the following identifies the correct procedure entry for current plant conditions AND the basis for entry?

A. 1210-1, to address the core cooling upset condition.

B. 1210-3, to address the over-cooling event.

C. 1210-5, to address challenge to RCS integrity.

D. 1210-6, to address the Small Break LOCA condition.

Technical Reference TDR No. 1197, pages 4 and 5, Rev 2.

Open Exam Reference

Learning Objective V.E.11.08 Given plant conditions, determine if entry into the ATPs is required and explain the proper entry point.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .10 ☐ 55.43 ☑ 55.45 .13

Discriminant Validity Statements

A Correct answer.

B Plausible since an over-cooling occurred.

C Plausible due to tube leak in progress.

D Plausible due to tube leak in progress.

Comments None.

Q-099

GPU NUCLEAR Technical Data Report	TDR No. 1197	Revision No. 2
	Budget Activity No. 115403	Page 1 of 38
Project: TMI-1	Department/Section: Nuclear Analysis & Fuels/Safety & Risk Analysis	
	Revision Date 11/11/99	

Document Title: Comparison of ATP 1210-1 and BWOG GEOG

Originator Signature	Date	Approval(s) Signature	Date
John Link	11/99		11/16/9
		Approval for External Distribution	Date

Does this TDR include recommendation(s)? [ ] Yes [x] No If yes, TFWR/TR # _____

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Abstract:

This report provides a comparison of the TMI-1 Abnormal Transient Procedure ATP 1210-1, "Reactor Trip", with the B&W Owner's Group Generic Emergency Operating Guidelines. Differences are identified and justification provided for each difference.

This is a report of work conducted by an individual(s) for use by GPU Nuclear, Inc. Neither GPU Nuclear, Inc. nor the authors of the report warrant that the report is complete or accurate. Nothing contained in the report establishes company policy or constitutes a commitment by GPU Nuclear, Inc.

* Abstract Only

TMI

1.0 **ENTRY CONDITION** - Any condition requiring an automatic or manual trip signal or a symptom of core cooling upset occurs while shutdown prior to DHR operation.

GEOG

"ENTRY STATUS"

- A reactor trip has occurred.
- Any RPS reactor trip limit or other plant specific limit requiring reactor trip is present whether or not the reactor has tripped.
- A symptom (Loss of SCM, Excessive Heat Transfer, Loss of Heat transfer, SGTR) has occurred while the reactor is shutdown above decay heat removal system operation.
- SGTR symptom occurs while reactor is not shutdown.

Discussion:

The TMI Entry Condition (Step 1.0) and the GEOG Section III.A Entry Status are basically the same. The SGTR symptom while the reactor is not shutdown is addressed by direct entry into ATP 1210-5. In the GEOG, this entry condition is addressed as a Forced Shutdown and immediately transfers to SGTR Section III.E step 1.0. The ATP configuration eliminates a procedure transfer by direct entry into ATP 1210-5. There is no technical difference and the mitigation strategy is not affected. OS-24, "ATOG Theory and Practice", states the following:

"4.1.2 Entry Points: The TMI Abnormal Transient Procedures have two entry points:

ATP 1210-1, Reactor Trip; Any condition requiring an automatic or manual trip signal or a symptom of core cooling upset occurs prior to DHR operation, (during RCS heatup and cooldown).

ATP 1210-5, OTSG Tube Leakage: Identified OTSG tube leakage greater than 1 gpm.

A separate entry point for OTSG Tube Leakage allows for better procedure flow when the reactor is at power when a SGTR occurs. The OTSG Tube Leakage procedure treats the symptoms of a tube leak, controls the plant shutdown and transfers to the Reactor Trip procedure to perform the Vital System Action, Status Verification, and Remedial Actions when a reactor trip occurs, or when the manual trip pushbutton is pressed. If a SGTR is the only symptom of core cooling upset present the operators should enter ATP 1210-5. If a symptom of core cooling upset of a higher priority occurs at the same time as a SGTR then the operator should enter ATP 1210-1. The highest priority symptoms of core

cooling upset must be addressed first. The symptoms of a SGTR cannot be treated correctly if core cooling upset conditions are not being treated."

Form ES-401-6

Written Exam Question Worksheet

Q # 100

Examination Outline Cross-Reference RO SRO

KA # Emerg Procedures/Plan 2.4.49 Page Tier # 3

RO/SRO Importance Rating 4.0 4.0 Group #

Measurement Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Proposed Question ☑ RO ☑ SRO ☑ PRA Related **Correct Answer** A

The plant was operating at 100% power with all systems in a normal line-up when the following events occurred:
- Small break LOCA requiring reactor trip and initiation of HPI
- SCM was lost and is not being recovered
- MU-P-1B tripped when HPI was actuated

The following plant conditions exist:
- RCS pressure is 385 psig and dropping at 10 psig per minute.
- Seal injection is being maintained at 34 gpm.
- Train 'A' HPI flow is 510 gpm
- Train 'B' HPI flow is 510 gpm

Based on current plant conditions, HPI flow should be:

A. Throttled using MU-V-16A and MU-V-16B on Train 'A'.

B. Throttled using MU-V-16C and MU-V-16D on Train 'B'.

C. Maintained as is until Low Pressure injection is adequate.

D. Maintained as is until subcooling margin is restored.

Technical Reference ATOG 1210-10, Abnormal Transient Rules, Guides, and Graphs, page 4, Rev. 37.

Open Exam Reference

Learning Objective IV.A.09.30 Explain how HPI flow is controlled and under what conditions it may be throttled.

Question Source ☑ New ☐ TMI Bank TMI Question #
☐ Modified TMI Bank Parent Question #

Question History ☐ Used in Last Two TMI NRC Exams Exam Date
☐ Used in Audit Exam ☐ Used in Training Program Quiz Date

Question Cognitive Level ☐ Memory or Fundamental Knowledge
☑ Comprehension or Analysis

10 CFR Part 55 Content ☑ 55.41 .10 ☑ 55.43 .2 ☑ 55.45 .6

Discriminant Validity Statements

A Correct answer. HPI flow must be throttled to prevent pump runout at 515 gpm.

B Plausible if examinee misinterprets seal injection flow source.

C Plausible since LPI flow is one of the conditions allowing throttling of HPI.

D Plausible, since SCM is one of the conditions allowing throttling of HPI.

Comments None.

FOR INFORMATION ONLY

GPU NUCLEAR	TMI Abnormal Transient Procedure	Number 1210-10
Title: **Abnormal Transients Rules, Guides and Graphs**		Revision No. 37
Applicability/Scope: TMI Division	Responsible Office: Plant Operations Dir.	Effective Date: 10/15/99

This document is within QA plan scope	X	Yes		No
Safety Reviews Required	X	Yes		No

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	Signature	Date
Originator	[illegible]	10-14-99
Procedure Owner	James R Paulles	10/15/99
PRG	[illegible]	10/15/99
Approver	James R Paulles	10/15/99

1.0 RULES

1.1 Subcooling Margin (SCM) Determination/Limit - Rule

The minimum subcooling margin shall be determined and limited as follows:

1.1.1 Reactor Coolant Pumps Operating or Natural Circulation Verified: The minimum SCM is 25°F as determined by the most conservative of the two subcooling margin meters (PCL).

1.1.2 Natural Circulation Not Verified:

a. The minimum SCM is 25°F as determined by the value of Comp. Pt. C4008 (calculated Incore SCM) or the average of the five (5) highest, operable incore thermocouples (Comp. Pt. C4006) and RCS wide range pressure PI-949A (as read on panel PCL) or PT-963 (Comp. Pt. A0404).

b. The minimum SCM is 25°F as determined by the highest operable BIRO incore thermocouple and RCS wide range pressure PI-949A (as read on panel PCL) or PT-963 (Comp. Pt. A0404).

1.2 High Pressure Injection (HPI) Initiation Criteria

Initiate two (2) HPI Pumps, full flow in ES Alignment when:

1.2.1 1600 psig ESAS has or should have auto initiated or

1.2.2 Subcooling margin is < 25°F or

1.2.3 Neither OTSG is available as a heat sink (Lack of Heat Transfer and neither Main FW nor EFW available)

1.3 High Pressure Injection (HPI) Throttling Criteria

Throttle HPI only if the following criteria are met:

1.3.1 HPI must be throttled to prevent pump runout (515 gpm/pump).

1.3.2 Do not throttle to < 500 gpm/pump unless one of the following three criteria is met.

a. HPI must be throttled to prevent violation of the RV P-T Limits Curve and to minimize SCM per PTS guidance.

OR b. HPI may be throttled if LPI flow is > 1000 gpm in each line AND Superheat does not exist.

OR c. HPI may be throttled if the required 25°F subcooling margin exists.

1.3.3 IF HPI is throttled per 1.3.2; THEN Open MU-V-36 & 37 as required to maintain MU pump flow > 100 GPM.

STEAM GENERATOR DEMAND
FROM DWG. D553730
TEMP. COMPENSATED F.W. FLOW "A"
FW FLOW RECORDER (SP9A-FR)
(+ TOTAL FLOW
FW FLOW DEMAND)
FEEDWATER ERROR
DWG. D553732
TO REACTOR CONTROL
FW TO Rx XLIMIT
SELECTED TEMP COMPENSATED REACTOR COOLANT FLOW "A" (A5027)
TAVE ERROR
(+ TEMP. > SET POINT)
AVERAGE TEMP. CONTROL
FROM REACTOR CONTROL
DWG. D553732
SET AT ZERO ERROR
SET AT ZERO ERROR
SET AT 95
SET AT 95
FEEDWATER TEMP. (A5043)
NEUTRON ERROR
FROM REACTOR CONTROL
DWG. D553732
TO FW XLIMIT
TEMP. COMPENSATED F.W. FLOW "B"
FW FLOW RECORDER (SP9B-FR)
SELECTED STEAM GEN. B PRESS. DWG. D554567 (A5058)
SELECTED TEMP. COMPENSATED REACTOR COOLANT FLOW "B" (A5056)
SELECTED REACTOR COOLANT TEMP. DWG. D554566
STEAM GEN. A PRESS. DWG. D554567
REACTOR INLET TEMP. LOOP A/B DWG. D554566 ΔTc
ΔTc CONTROL
BTU LIMITS
BTU LIMITS
BTU LIMIT
BTU LIMIT
RC FLOW ERROR
FW ERROR
FW ERROR
TOTAL FW DEMAND
S.G. "B" LEVEL (0-100%) (SP9B-LT-1288) DWG. D554567 (A5049)
S.G. "A" LEVEL (0-100%) (SP9A-LT-1288) (A5047)
S.G. LEVEL "B" (START-UP) (0-365") (A5050)
S.G. LEVEL "A" (START-UP) (0-365") (A5048)
HI LIMIT
LO LIMIT
OP LEVEL ERROR
SU LEVEL ERROR
SU LEVEL ERROR
VALVE "A"
VALVE "B"
SELECTED F.W. VALVE ΔP
"B" START-UP F.W. VALVE
"B" MAIN F.W. VALVE
"B" F.W. PUMP TURB. SPEED CHANGER (A5032)
"A" F.W. PUMP TURB. SPEED CHANGER (A5030)
"A" MAIN F.W. VALVE
"A" START-UP F.W. VALVE
GPU NUCLEAR
ADMINISTRATIVE REVISION
REV DRAFT CHECK APPROVED DATE APP
REVISION
CAD FILE: D553731_RQ
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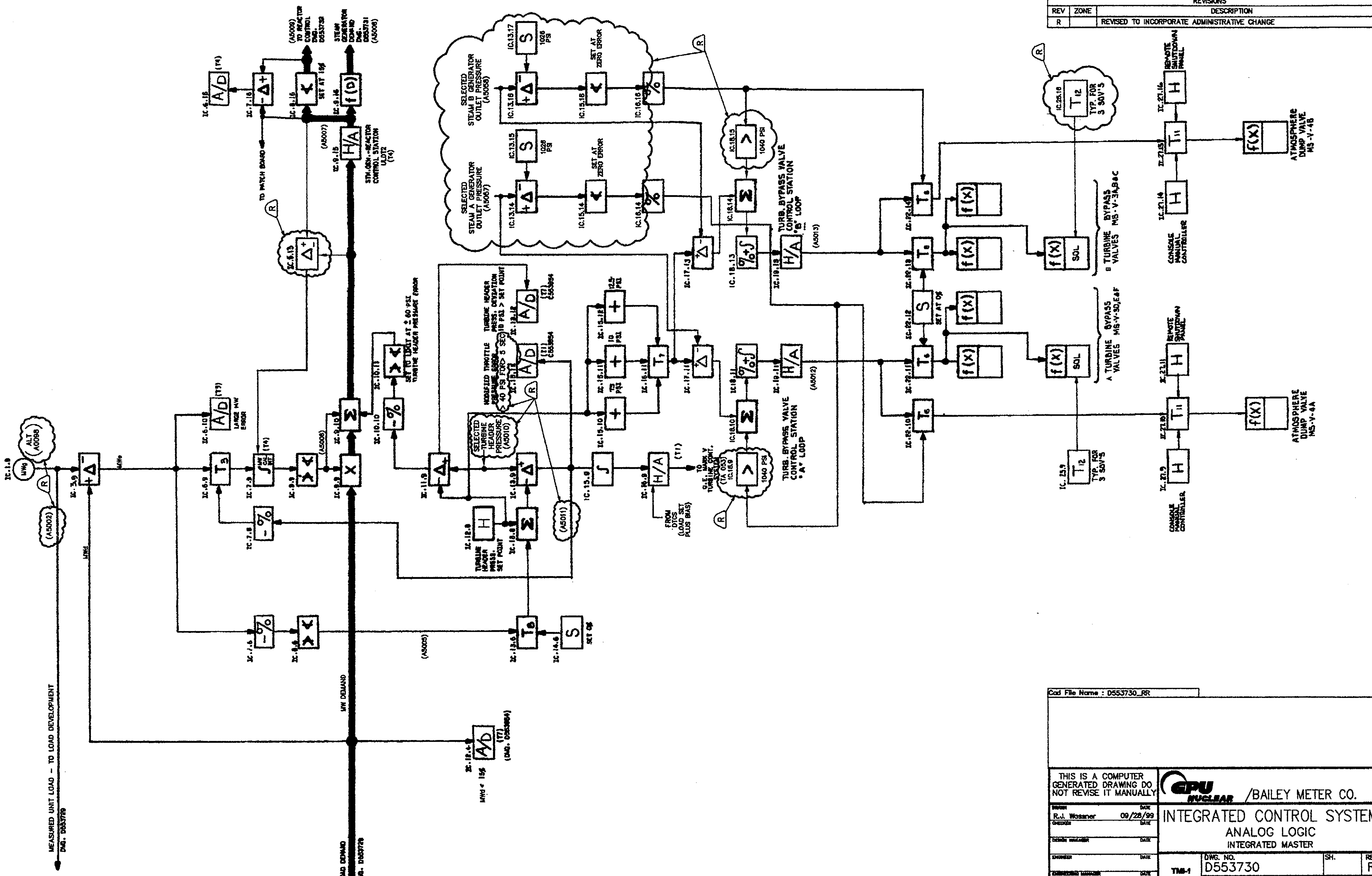
REV. PER. #/P (SEE)
REDRAW IV
CHANGE
BABCOCK & WILCOX DRAWING NUMBER 520-0005 -30-30K-07
INTEGRATED CONTROL SYSTEM
ANALOG LOGIC
FEEDWATER CONTROL
CUSTOMER: METROPOLITAN EDISON CO.
PLANT: THREE MILE ISLAND NUCLEAR STATION
CONTRACTING ENG: B&W
B&W CO. JOB NO: 520
BW ORDER NO.: W.A.-17-92
CUST. ORDER NO.: 520-0005
DATE 2/18/69
D 553731
Q

DWG.NO. D553730 SH. -1

REVISIONS		
REV	ZONE	DESCRIPTION
R		REVISED TO INCORPORATE ADMINISTRATIVE CHANGE



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REFERENCES		

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DRAWN R.J. Wassner DATE 09/28/99
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ENGINEER DATE
ENGINEERING MANAGER DATE

INTEGRATED CONTROL SYSTEM
ANALOG LOGIC
INTEGRATED MASTER

TMI-1 DWG. NO. D553730 SH. REV. R
SCALE: NONE

INTERFACING CONCURRENCE

BA NO. –