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REGION 1

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AmerGen
An Exelon/British Energy Company

May 23, 2003

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

Subject: Submittal of Senior Reactor Operator Final Examination Materials
Three Mile Island Unit 1 (Docket # 50-289)

In accordance with NUREG 1021, Revision 8, "Operating Licensing Examination Standards for Power Reactors", AmerGen is submitting the final Senior Reactor Operator examination materials for Three Mile Island Unit 1.

This submittal provides final examination documentation for the simulator and JPM examinations administered by the NRC on May 12, 13, and 14, as well as the written examination administered by AmerGen for the NRC on May 19, 2003.

Should you have any questions concerning this letter or the examination materials, please contact Dennis Boltz at (717) 948-2006.

Respectfully,

Dennis J. Boltz

Dennis J. Boltz
Exam Author
Three Mile Island Unit 1

Enclosures: (Sent directly to Joseph D'Antonio, Chief Examiner, NRC Region I)

Written Examination:
- Master Examination and Answer Key.
- Examination Answer Sheets – originals and ungraded.
- Exam Comment Sheet - describing proctor clarifications during the written exam.
- Examinee Seating Arrangement.
- Exam/Answer Key Adjustments, with supporting documentation.
- AmerGen Written Exam Grading Results.
- Written Examination Performance Analysis.
- ES-403-1, Written Examination Grading Quality Checklist.
- ES-201-3 Examination Security Agreement.

Dynamic Simulator Scenarios
- Day 1 and Day 2.

JPMs:
- Administrative, Facility Walkthrough, and In-Plant.

Exelon Nuclear

EXAM COMMENT SHEET

DATE: May 19, 2003 SUBMITTED BY: Dennis J. Boltz RO/**SRO**

EXAM: WRITTEN / WALK-THROUGH / SIMULATOR SCENARIO (circle one)

Test Item (Question/JPM/Scenario, etc.)	Concern or Problem	Recommended Resolution	Reference	Remarks
Q-005	Examinee asked for clarification of location of "Breaker Trip Lights."	No change for NRC Exam. Consider editing this question prior to entering this question into the TMI exam bank.	N/A.	Clarification provided: Breaker trip lights addressed in the question stem are located on outside front of the RPS Cabinets, at the top right corner.
Q-018	Examinee asked if intent of question was to identify the immediate automatic response.	No change for NRC Exam. Consider editing the question prior to entering this question into the TMI exam bank.	N/A.	Proctor response: Yes, the intent of the question is to address the immediate automatic response.
Q-075	Proctor noticed typographical error in answer A – RM-A-4 should be RM-A-6.	No change for NRC Exam. Correct typographical error prior to entering this question into the TMI exam bank.	NA	Announced and noted the typographical error and the correction on the white board in front of the exam room.

Additional comments: None

Exam Analyzer comments: None – Dennis J. Boltz

Final Resolution:

Reviewed by: Dennis J. Boltz
Facility Author

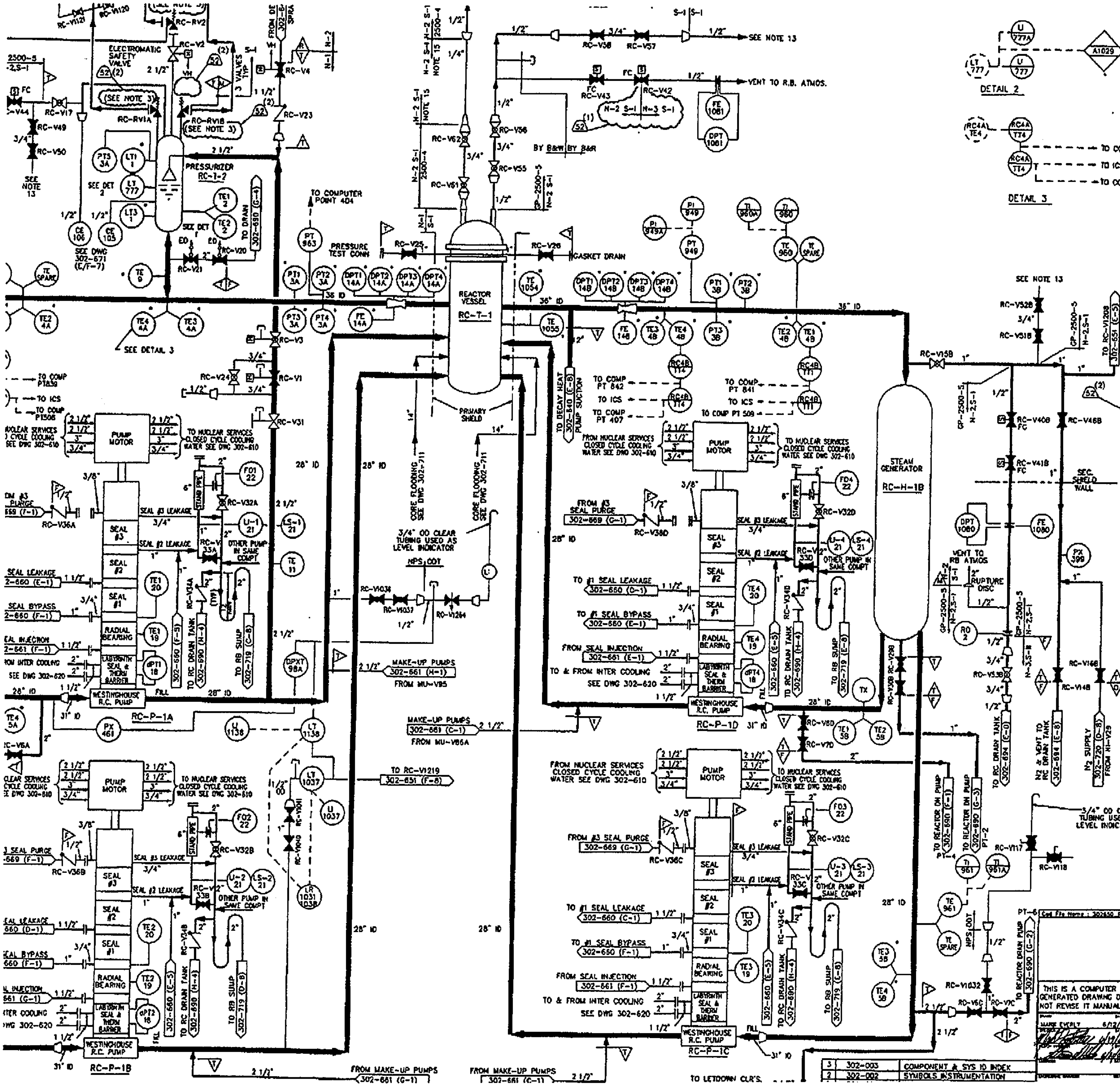
Approved by: Dennis J. Boltz
Facility Representative

May 19, 2003
NRC SRO Initial License Written Exam
Exam/Answer Key Adjustments

Q	Comment	Resolution	Reference
26	Answer Key identifies D as the correct answer, stating that Pressurizer system failure has occurred. Answer B can also be a credible answer if loss of SI/MU occurs. In order to select answer D, the examinee must assume the Pressurizer system failure. Given that the Loss of MU/SI procedure is included in 3 of the 4 answer choices, assuming loss of MU/SI has occurred is not unreasonable.	**Accept both B and D as correct answers.** With loss of RCP seal injection, seal #2 leak-off would provide a transport flow path to carry high activity RCS water to the RC Drain Tank, even in the absence of a Pressurizer system failure. Even though this bank question was previously approved and used in the 2001 TMI NRC SRO written exam as written, it will be edited in the TMI question bank to preclude 2 correct answers for future use.	**Attachment 1** RCS P&ID 302-650 shows flow path from Seal #2 leak off to the RC Drain Tank.
42	As written, there is no correct answer provided.	**Delete this question** IAW ES-403 Section D.1.b, since there is no correct answer provided. The intent of question was to have two Pressurizer level channels inoperable in excess of 48 hours, to require a Tech Spec shutdown. As written, LI-777 is still operable. Therefore there is no correct answer given. This question will be edited prior to entry into the TMI question bank to represent conditions with both level channels inoperable.	**Attachment 2** TS 3.5.5, Accident Monitoring Instrumentation, and bases, pages 3-40a, 3-40b, and 3-40c. Page 3-40b states LT-1 and LT-3 are common to ONE channel, and that LT-777 is the other channel.
43	Based on the OS-24 definition for "Available as a Heat Sink," Rule 4, FW Control, does not apply.	**Change correct answer from A to D.** Since OS-24 states that a dry OTSG is not considered available as a heat sink, Rule 4, FW Control, does not apply to the conditions in the question stem. Therefore, Guide 13, Dry OTSG, applies. Edit this question prior to entry into TMI question bank: stem should state that both OTSGs are NOT dry.	**Attachment 3** OS-24, Conduct of Operations During Abnormal and Emergency Events, Rev 7, Section 3.9. OP-TM-EOP-010, Abnormal Transients Rules, Guides and Graphs, Rev. 1 – Rule 4, FW Control, and Guide 13, Dry OTSG.
74	Answer Key is not correct.	**Change Answer Key from B to C.** This question was apparently edited, and then answers B and C were interchanged for psychometric considerations. This change brings the Answer Key into conformance with the NRC approved question and discriminant validity statements.	**Attachment 4** Form ES-401-6 for Question #74, as approved by the NRC during week of April 7, 2003, Examiner prep week.

This is what is in ADAMS ML 03170 0056?

See 2nd CLIP - Where is that letter?



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 TO BE FOUND IN THE APPLICABLE ENGINEERING SETPOINT DATABASE.
4. B & W FURNISHED INSTRUMENTATION MARKED WITH AN ASTRISK (*).
5. SEE B & W DWG 135756E FOR PROCESS AND INSTRUMENTATION SYMBOLS AND IDENTIFICATION.
6. FOR DESIGN CONDITIONS AND B & W INST SEE B & W DWG 26553F.
7. FOR B & W FURNISHED PIPING SEE B & W DWGS 131942E, 131943E, 135841E & 135842E.
8. ALL PIPING IS SEISMIC CLASS I & NUCLEAR CLASS I EXCEPT AS NOTED.
9. ALL PIPING IS SEISMIC CLASS II & NUCLEAR CLASS III.
10. ALL PIPING IS SEISMIC CLASS II & NUCLEAR CLASS II EXCEPT AS NOTED.
11. ALL PIPING IS SEISMIC CLASS I & NUCLEAR CLASS I EXCEPT 1" AND SMALLER LINES WHICH ARE SEISMIC CLASS I & NUCLEAR CLASS II.
12. HIGH POINT OF LOOP SEAL TO BE 12" ABOVE HIGH POINT OF PIPE RUN TO R.C. DRAIN TANK.
13. TEST CONN'S. ARE PROVIDED FOR HYDRO TESTING AND/OR FLOW PATH INTEGRITY TEST OF VENT PIPING.
14. ALL PIPING IS SEISMIC CLASS III & NUCLEAR CLASS 3.
15. INSTRUMENT IMPULSE LINE TUBING IS PER GPUN SPEC. SP-900044-001 SMLS. STL. UNLESS OTHERWISE NOTED.

Q26 Reference

Attachment 1

3	302-003	COMPONENT & SYS ID INDEX
2	302-002	SYMBOLS INSTRUMENTATION

THIS IS A COMPUTER GENERATED DRAWING DO NOT REVISE IT MANUALLY

MARK EVERLY 6/12/01

AmerGen

REACTOR COOLANT SYSTEM

FLOW DIAGRAM

TMI-1 DWG. NO. 302-650 REV. 52

Q-0412 Reference
Attachment 2

3.5.5 ACCIDENT MONITORING INSTRUMENTATION

Applicability

Applies to the operability requirements for the instruments identified in Table 3.5-2 and Table 3.5-3 during STARTUP, POWER OPERATION and HOT STANDBY.

Objectives

To assure operability of key instrumentation useful in diagnosing situations which could represent or lead to inadequate core cooling or evaluate and predict the course of accidents beyond the design basis.

Specification

3.5.5.1 The minimum number of channels identified for the instruments in Table 3.5-2, shall be OPERABLE. With the number of instrumentation channels less than the minimum required, restore the inoperable channel(s) to OPERABLE status within seven (7) days (48 hours for pressurizer level) or be in at least HOT SHUTDOWN within the next six (6) hours and in COLD SHUTDOWN within an additional 30 hours. Prior to startup following a COLD SHUTDOWN, the minimum number of channels shown in Table 3.5-2 shall be operable.

3.5.5.2 The channels identified for the instruments specified in Table 3.5-3 shall be OPERABLE. With the number of instrumentation channels less than required, restore the inoperable channel(s) to OPERABLE in accordance with the action specified in Table 3.5-3.

Bases

The Saturation Margin Monitor provides a quick and reliable means for determination of saturation temperature margins. Hand calculation of saturation pressure and saturation temperature margins can be easily and quickly performed as an alternate indication for the Saturation Margin Monitors.

Discharge flow from the two (2) pressurizer code safety valves and the PORV is measured by differential pressure transmitters connected across elbow taps downstream of each valve. A delta-pressure indication from each pressure transmitter is available in the control room to indicate code safety or relief valve line flow. An alarm is also provided in the control room to indicate that discharge from a pressurizer code safety or relief valve is occurring. In addition, an acoustic monitor is provided to detect flow in the PORV discharge line. An alarm is provided in the control room for the acoustic monitor.

Q-042

3.5.5 ACCIDENT MONITORING INSTRUMENTATION (Continued)

The Emergency Feedwater System (EFW) is provided with two channels of flow instrumentation on each of the two discharge lines. Local flow indication is also available for the EFW System.

Although the pressurizer has multiple level indications, the separate indications are selectable via a switch for display on a single display. Pressurizer level, however, can also be determined via the patch panel and the computer log. In addition, a second channel of pressurizer level indication is available independent of the NNI.

Although the instruments identified in Table 3.5-2 are significant in diagnosing situations which could lead to inadequate core cooling, loss of any one of the instruments in Table 3.5-2 would not prevent continued, safe, reactor operation. Therefore, operation is justified for up to 7 days (48 hours for pressurizer level). Alternate indications are available for Saturation Margin Monitors using hand calculations, the PORV/Safety Valve position monitors using discharge line thermocouple and Reactor Coolant Drain Tank indications, and for EFW flow using Steam Generator level and EFW Pump discharge pressure. Pressurizer level has two channels, one channel from NNI (2 D/P instrument strings through a single indicator) and one channel independent of the NNI. Operation with the above pressurizer level channels out of service is permitted for up to 48 hours. Alternate indication would be available through the plant computer.

The operability of design basis accident monitoring instrumentation as identified in Table 3.5-3, ensures that sufficient information is available on selected plant parameters to monitor and assess the variables following an accident. (This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," Rev. 3, May 1983.) These instruments will be maintained for that purpose.

Q-242

TABLE 3.5-2

ACCIDENT MONITORING INSTRUMENTS

FUNCTION	INSTRUMENTS	NUMBER OF CHANNELS	MINIMUM NUMBER OF CHANNELS
1	Saturation Margin Monitor	2	1
2	Safety Valve Differential Pressure Monitor	1 per discharge line	1 per discharge line
3	PORV Position Monitor	2	1*
4	Emergency Feedwater Flow	2 per OTSG	1 per OTSG
5	Pressurizer Level	2	1
6	Backup Incore Thermocouple Display Channel	4 thermocouples/core quadrant	2 thermocouples/core quadrant

* With the PORV Block Valve closed in accordance with Specification 3.1.12.4.a, the minimum number of channels is zero.

3.7 LACK OF PRIMARY-TO-SECONDARY HEAT TRANSFER (LOHT) is the inability of either OTSG to remove sensible heat from the RCS. LOHT can be confirmed if :

- Neither OTSG has water level control and pressure control.
  AND
- Core exit temperatures are rising

3.8 MINIMIZE SCM: An intentional reduction of the reactor coolant pressure temperature relationship as close as practical to the 25°F subcooling margin or emergency RCP NPSH limit. (Recommended band 30-70°F)

3.9 OTSG AVAILABLE AS A HEAT SINK:

A physical condition where the OTSG demonstrates level and pressure control, used to determine if primary to secondary heat transfer is possible. (i.e. heat sink) Primary to secondary heat transfer need not be demonstrated to determine this availability. Primary to secondary leakage should not be considered a means of OTSG level control. A dry OTSG is not considered available as a heat sink.

3.10 OVERSIGHT:

The independent monitoring of plant and crew performance and <u>any</u> subsequent intervention, as needed, to ensure the appropriate mitigation strategy is being pursued for the current plant conditions. Refer to Attachment B, SM Oversight Management Guidelines.

3.11 "PLANNED" REACTOR TRIP

A scheduled shutdown, where a reactor trip, is directed by an approved procedure.

3.12 PRIMARY-TO-SECONDARY HEAT TRANSFER (PSHT) is the removal of sensible heat from the RCS to one or both OTSG(s). PSHT can be confirmed if:

- Either OTSG has water level control and pressure control.
  AND
- RCS T_c is approximately the same as secondary T_{sat} and responds to changes in OTSG pressure.
  AND
- RCS forced or verified natural circulation is present.

3.13 RCP AVAILABLE – An available RC Pump is one which can be operated without extraordinary efforts. Pump service functions (motor cooling, seal cooling, etc.) are operable (redundancy not required) and all interlocks can be satisfied. Strict compliance with administrative shutdown criteria (vibration, seal leakoff flow, etc) is not expected when the operation of the pump is more important to safe plant operation.

FWC

4

Rule 4
Feedwater Control

A. **IAAT** the reactor is shutdown, **then:**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
VERIFY SCM > 25°F.	**MAINTAIN** OTSG level 75 – 85% OPERATING Range Level.
VERIFY at least 1 RCP operating.	**MAINTAIN** OTSG level ≥ 50% OPERATING Range Level.
MAINTAIN OTSG level ≥ 25" STARTUP Range Level.	

B. **If** Level < minimum, **then MAINTAIN** the following MINIMUM required flow:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
If SCM < 25°F **and** both OTSGs are available, **then FEED** > 215 gpm/OTSG using EFW.	**If** EFW is not available, **then FEED** > 1.0 Mlbm/hr using MFW.
If SCM < 25°F **and** only one OTSG is available, **then FEED** > 430 GPM to the good OTSG using EFW.	**If** EFW is not available, **then FEED** > 1.0 Mlbm/hr using MFW.
If RCPs are OFF, **then FEED** OTSG at maximum available using EFW, within RCS Cooldown rate limit.	**If** EFW is not available, **then FEED** > 1.0 Mlbm/hr using MFW.
There is no minimum required flow rate.	

Guide 13
Dry OTSG

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1. **VERIFY** OTSG SU Level < 6" **and** OTSG pressure at least 200 psi below P_{sat} for T_c.	
2. **MONITOR** Tube to Shell Differential Temperature (TSDT) and **REVIEW** Guide 14.	
3. **VERIFY** the other OTSG is available.	**GO TO** Step 5.
4. **VERIFY** all RCPs are OFF **or** TSDT Limits are being challenged	

NOTE

Automatic EFW actuation is not restricted by this guidance. Limit feedwater flow to the Dry OTSG until OTSG pressure has been restored. RCP operation is desired.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5. **VERIFY** the DRY OTSG pressure boundary is INTACT.	**VERIFY** the OTSG pressure boundary failure is **<u>not</u>** in the Intermediate or Reactor Building.
6. **If** TSDT tensile limit is being challenged, **then,** 1) **If** OTSG pressure boundary is **<u>not</u>** intact, **then VERIFY** an RCP is operating. 2) **FEED** the DRY OTSG at a maximum flow of 0.1 Mlb/HR using Main Feedwater.	**If** RCPs are OFF, **then FEED** the DRY OTSG at a maximum of 185 GPM using EFW.
7. **If** TSDT compressive limit is being challenged, **then,** 1) **If** <u>at least one</u> RCP is ON, **then FEED** the DRY OTSG at a maximum of 435 GPM using EFW.	**If** RCPs are OFF, **then FEED** the DRY OTSG at a maximum of 185 GPM using EFW.

Attachment 4

Form ES-401-6 **Written Exam Question Worksheet** Q # 074

Examination Outline Cross-Reference

					RO	SRO
SYS/EP# 068	KA # K1.07	Page # 3.9-2	Tier #			2
RO/SRO Importance Rating	2.7	2.9	Group #			1

Measurement Knowledge of the physical connections and/or cause-effect relationships between Liquid Radwaste System (LRS) and the following systems: Sources of liquid wastes for LRS.

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

Plant conditions:
- Reactor power is 100%, with ICS in full automatic.

Based on these conditions, identify the ONE selection below that describes a NORMAL source of water to the Liquid Waste Disposal System.

A. PORV pilot valve leakoff.

B. Leakoff from between the reactor vessel flange O-Rings.

C. Intermittent drain flow from the Waste Gas Compressor Separator.

D. Valve packing leakage from Letdown isolation valves MU-V-1A and MU-V-1B.

Technical Reference 302-696, Waste Gas Compressors Flow Diagram, Rev. 1.

Open Exam Reference None.

Learning Objective IV.B.09.01

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

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 Incorrect answer.

B Incorrect answer. This line is normally isolated.

C Correct answer. Level switch operates solenoid operated valve to drain excess water from the separator to the Auxiliary Building Sump.

D Incorrect answer. Packing leakoff lines are capped.

Comments None.

TMI May 2003 SRO NRC Written Examination
Examination Review
Individual Remediation
Examinee Individual and Group Performance Analysis

Immediately following completion of the written examination by all the examinees, every test question was reviewed with all the examinees to obtain comments and feedback regarding question and answer key validity, and to accomplish immediate individual remediation for all incorrect examinee responses. Refer to the documentation provided that describes answer key adjustments with supporting justifications.

In addition, a written examination error analysis spreadsheet was developed to facilitate reviews of individual and group performance on the examination. All incorrect answers were evaluated independently by three SRO certified people with extensive training experience, knowledgeable in TMI systems, operating, administrative, abnormal and emergency procedures, and station operating license requirements.

The results of these reviews are as follows:

- No trends were identified that suggest the existence of a common group weakness that requires training material upgrades or more emphasis in future programs.
- No areas of individual weakness were identified that require remediation training in addition to that already conducted for individual examinees.

Please refer to the attached TMI Written Examination Error Analysis spreadsheet for a detailed presentation of all incorrect answers.

TMI Written Exam
Error Analysis

TMI SRO Initial Exam
May 19, 2003

Q	Noble	Smith	Saltz	Parfitt	Composite Results	KA Description
	Incorrect Responses					
4				B	B	Knowledge of the operational implications of axial power imbalance as applicable to Inoperable/Stuck Control Rod.
5	C		D		CD	Knowledge of the interrelations between Reactor Trip and the Reactor trip status panel.
7		D		B	BD	Knowledge of the interrelations between Pressurizer Vapor Space Accident (Relief Valve Stuck Open) and Valves.
10	C		C		CC	Knowledge of the operational implications of natural circulation cooling, including reflux boiling, as applied to Small Break LOCA.
21			B		B	Knowledge of the reasons for termination of startup following loss of IR NIs.
25	C	C		C	CCC	Knowledge of the reasons for length of time battery capacity is designed as applied to Station Blackout.
26			C		C	Ability to determine and interpret ARM panel displays.
30	A				A	Knowledge of the operational implications of RB pressure on leakrate as applied to loss of containment integrity.
31			C	C	CC	Ability to determine and interpret changes in Pressurizer level due to steam bubble transfer to the RCS during inadequate core cooling.
39		B			B	Knowledge of the operational implications of annunciators and condition indicating signals, and remedial actions associated with a LOCA Cooldown.
41		C	B		BC	Knowledge of the reasons for the normal, abnormal and emergency operating procedures associated with natural circulation cooldown.
43		A		A	AA	Knowledge of the operational implications of normal, abnormal and emergency operating procedures associated with EOP Rules.
44	C		D		CD	Knowledge of the effect of a loss or malfunction of reactor trip breakers, including controls, on the Control Rod Drive System.
46	C			B	BC	Knowledge of the operational implications of brittle fracture as applied to the RCS.
57	A				A	Knowledge of the effect of a loss or malfunction of trip setpoint calculators on RPS.
58	A	A	B		AAB	Knowledge of Nuclear Instrumentation System design feature(s) and/or interlock(s) which provide for slow response time of SPNDs.
60				D	D	Knowledge of bus power supplies to the following: Containment Cooling Fans.
64	D	D			DD	Ability to monitor automatic operation of the Spent Fuel Pool Cooling System.
65		D			D	Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling Equipment System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: dropped cask.

TMI Written Exam
Error Analysis

68			C	
78	C		C	
83	B			
90	D	A		
92		C		
96		D		D
97		B	B	
100	C			B

C	Ability to monitor automatic operation of the Condenser Air Removal system, including automatic diversion of exhaust.
CC	Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Area Radiation Monitoring (ARM) System controls including: Radiation levels.
B	Knowledge of which events related to system operations/status should be reported to outside agencies: Containment System.
AD	Knowledge of the process for controlling temporary changes. (Equipment Control).
C	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity. (Equipment Control).
DD	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure. (Radiation Control).
BB	Knowledge of the process used track inoperable alarms. (Emergency Procedures/Plan).
BC	Knowledge of annunciator response procedures.(Emergency Procedures/Plan).