U.S. NUCLEAR REGULATORY COMMISSION
REGION I
OPERATOR LICENSING EXAMINATION REPORT

Examination Report No. 87-25 (OL)

Facility Docket No. 50-289

Facility License No. DPR-50

Licensee: GPU Nuclear Corporation
P.O. Box 480
Middletown, Pennslyvania 17057

Facility Name: Three Mile Island, Unit 1

Examination Dates: December 8-11, 1987

Chief Examiner N. F. Dudley — 2-8-88
N. Dudley, Senior Operations Engineer — date

Approved By: R M Keller — 2/10/88
Robert M. Keller, Chief, PWR Section — date

Summary: Written examinations and operating tests were administered to two (2) senior reactor operator and five (5) reactor operator candidates. One reactor operator candidate did not pass the written examination. All other candidates passed the examinations.

Numerous individual deficiencies were noted during the operating test which may be indicative of a weakness in the training program.

## REPORT DETAILS

TYPE OF EXAMS: Replacement

EXAM RESULTS:

| | RO Pass/Fail | SRO Pass/Fail |
|---|---|---|
| Written Exam | 4/1 | 2/0 |
| Operating Exam | 5/0 | 2/0 |
| Overall | 4/1 | 2/0 |

CHIEF EXAMINER AT SITE: N. Dudley, Senior Operations Engineer

OTHER EXAMINERS: B. Picker, EG&G
M. King, EG&G

1. The following is a summary of generic strengths or deficiencies noted on operating tests. This information is being provided to aid the licensee in upgrading license and requalification training programs. No licensee response is required.

   Strength: Due to the limited number of contaminated areas within the auxiliary building, supervisory and operations personnel are allowed access to most equipment without the requirement of donning protective clothing.

   Weaknesses: Candidates did not identify the root cause of plant transients which in most cases were instrument failures.

   The simulator has not incorporated major plant modifications such as the Heat Sink Protection System.

   The simulator malfunction cause and effect book did not clearly define the ramping time or severity of some faults such as steam line break and fuel element failure.

2. The following is a summary of generic strengths or deficiencies noted from the writing and grading of the written examinations. This information is being provided to aid the licensee in upgrading license and requalification training programs. No licensee response is required.

Strength: Learning objectives provided at the beginning of each chapter of the Operations Plant Manual were adequately written and defined the information in the chapter for which the candidates were held responsible.

Weaknesses: Operations Plant Manuals were not up to date. For example, the chapter on the OSTG did not reflect the instrumentation which has been added by the Heat Sink Protection System.

Operations Plant Manuals are not useful for understanding integrated systems response nor for initially understanding system operations. The Operations Plant Manuals are useful as a source of technical details of isolated system information.

Senior Reactor Operator candidates were not familiar with the details of radiological controls procedures. For example: the requirements for initiating liquid waste release (question 8.01) and the title of the person to whom a radiological deficiency report should be forwarded (question 8.07.b)

Reactor Operator candidates showed knowledge weaknesses in the following areas.

- Interlock between the Reactor Building Cooling Pumps and EF-V-4 and 5. (Question 2.076)
- The effect of the Thot signal on other systems and components. (Question 2.10)
- The effect on BTU limits of varying input parameters. (Question 3.02)
- The procedures for correcting log entries. (Question 4.11)
- The effect of the Heat Sink Protection System on feedwater valves during a steam line rupture. (Question 4.13)

3. Exit Interview:

NRC Personnel present: N. Dudley, Senior Operations Engineer

Number Facility Personnel present:

T. Broughton, TMI-1 Operations and Maintenance Director
D. Hassler, Licensing Engineer
W. Thompson, TMI-1 Operator Training Manager
L. Ritter, TMI-1 Plant Operations Administrator

Summary of comments:

The number of types of examinaticns administered during the week was summarized by the examiner. The examiner noted the ease of access through security check points and the cleanliness of the radiological controlled areas. The examiner thanked the training staff for their assistance in operating the simulator during the evaluations.

The examiner stated that an attempt would be made to provide examinations results within 30 days. The examiner stated that discussions of individual candidate weaknesses had been held with the licensee training staff and that questions on plant equipment problems had been discussed with the licensee operations staff and the resident inspector.

Attachments:

1. Written Examination and Answer Key (SRO)
2. Written Examination and Answer Key (RO)
3. Facility Comments on Written Examinations made after Exam Review with NRC Response to Facility Comments.

ENCLOSURE 3

TMI REACTOR OPERATOR EXAMINATION
RESOLUTION OF COMMENTS OF EXAMINATION ADMINISTERED ON 12/8/87

FACILITY COMMENT:
ANSWER 1.07

A. Correct answer should be HIGHER

The point of adding heat (POAH) will be higher if temperature is higher. If the ambient temperature is higher than some reference condition, it will require higher power generation for the heat input from fission to raise fuel temperature significantly. Also the doppler coefficient becomes less negative as temperature increases.

Reference: OPM section N-7 page 70, item F.4.C
page 100, item I.2
page 101, figure 43

NRC RESOLUTION:

Comment Accepted. Answer will be changed to HIGHER.

FACILITY COMMENT:
ANSWER 1.12

Also acceptable response should be:

Reactor Coolant Flow
Further away from minimum DNBR

Reference: OPM section N-2, pages 177, 178 & 179

NRC RESOLUTION:

Comment Accepted. Answer key will be modified to accept Reactor Coolant Flow with minimum DNBR being further away.

FACILITY COMMENT:
ANSWER 1.13

The correct answer for "Effective" is as follows:

Effective because as one group is reaching the top of its travel, it is moving through a low flux region of the core, the next group is beginning to travel also in a low flux region. The two groups moving through a low flux region has approximately the same worth as one group moving through a high flux region.

Reference: OPM section N-7 page 161, item J.7
Tech Spec section 3.5.2.7 bases,
pages 3-35, 35a, 35b & 36 (Note paragraph of interest on top of page 3-36)

NRC RESOLUTION:

Comment Accepted. Answer key will be changed to reflect that both groups are moving through an area of lower flux resulting in nearly the same worth as one group moving in a high flux area.

FACILITY COMMENT:
ANSWER 1.14

A. Acceptable answer should be as follows:

$$\frac{(30\ \text{gal/inch})\ (10\ \text{inches})\ (1.6\ \text{hot vol. to cold vol.})}{(24\ \text{gal/inch})} = 20\ \text{in.}$$

OR

$$(30\ \text{gal/inch})\ (10\ \text{inches}) \ / \ (15\ \text{gal/inch}) = 20\ \text{in.}$$

The physical size of the pressurizer is such that one inch of level is equal to 24 gallons. The rule of thumb for the pressurizer is that 15 gallons of cold (room temperature) water will expand at RCS temperature to fill those 24 gallons. (Note that 24/1.6 = 15)

Reference: OPM section B-1 page 14, figure 4

NRC RESOLUTION:

Comment Accepted. Answer Key will be modified to show an answer of 20 inches.

FACILITY COMMENT:
ANSWER 1.16

The new Inconel APSRs are LESS capable of controlling imbalance than the old APSRs.

The correct response to the question is than as follows:

Because of "gray" poison, which extends over longer portions of the APSR, imbalance response is more linear with rod motion and there is no "foldover" at low APSR position. (Foldover is defined as driving imbalance less positive by inserting APSRs)

Reference: Lesson Plan 11.2.01.309, Objectives E & G

NRC RESOLUTION:

Comment Accepted. Second sentence of answer changed to state: "The APSRs imbalance response is more linear and there is no "foldover."" [0.5]

FACILITY COMMENT:
ANSWER 2.03

C. Should be "4" psig RB not 3 psig as stated in the key.

Reference: OPM section F-6 page 8

D. Should be "30" psig RB not 10 psig as stated in the key.

Reference: OPM section F-6 page 8

NRC RESOLUTION:

Comments Accepted for both 2.03.C and D. Answer Key corrected.

FACILITY COMMENT:
ANSWER 2.04

Clarification to the answer key is suggested for the following:

- "Waste Liquid" refers to isolation of the RB sump and RCDT and should be acceptable as an alternate response.
  (Valves WDL-V-303,304,534 & 535 close)
- "Waste Gas" refers to isolation of the Vent Header to the RB, which connects to the RCDT and should be acceptable as an alternate response.
  (Valves WDG-V-3 & 4 Close)

Also the following other systems not included in the answer key are systems isolated from the RB:

- "RMS" which refers to the isolation of RM-A-2 by the closing of valves CM-V-1,2,3 & 4.
- "RB Normal Cooling" which refers to the closing of RB-V-2A and RB-V-7.

Reference: OPM section F-6 page 44

NRC RESOLUTION:

Comment noted and Accepted as noted. The request for alternate answer for waste liquid and waste gas will not be accepted because they are referring to individual components and not to the system. Additional systems will be added to the key for "RMS/Containment Radiation Monitoring" and "RB Normal Cooling".

FACILITY COMMENT:
ANSWER 2.09

A. Acceptable alternate response should be "Permit the core to be rapidly reflooded following a postulated cold line break".

B. The flow path description should be as follows, "Steam generated in the core flows up and through the internal vent valves and flows out to the leak".

Reference: OPM section B-3 page 10

NRC RESOLUTION:

Comments Accepted. Both answers A. and B. will be modified to allow the above as alternate answers.

FACILITY COMMENT:
ANSWER 2.10

Response #2 states "ICS" as an affected item, a more specific answer of BTU Limits" should also be acceptable.

Reference: OPM section F-3 page 49, figure 12 and page 54

NRC RESOLUTION:

Comment partially Accepted. The BTU limits will be accepted but the ICS answer will be stricken.

FACILITY COMMENT:
ANSWER 2.11

The answer key correctly states that MU-V-2A/2B would close, but those valves isolate the letdown flow path from the RCS not the RCP seal return. This is done to minimize the activity in the auxiliary building as the answer key states.

Reference: OPM section B-5 page 25 & 26, item E.9
page 49, figure 6

NRC RESOLUTION:

Comment Accepted. Answer key modified to state the closure of the MU-V-2A/2B isolates letdown.

FACILITY COMMENT:
ANSWER 2.14

If the candidate assumed a 1600# ESAS actuation the following valves will reposition and pumps receive a start signal:

-MU-V-2A/2B
-MU-V-14A/14B
-MU-P-1A/1C
-MU-V-36/37
-MU-V-16A/16B/16C/16D
-MU-V-1B

If the candidate assumed a 4# ESAS actuation the following additional valve would repositon:

-MU-V-3

Reference: OPM section P-6 page 44

NRC RESOLUTION:

Comment Noted. MU-V-3 is already part of the answer key.

FACILITY COMMENT:
ANSWER 3.06

A. The question asks "What is the function..." and the proper response should be "The function is to isolate the affected components from their circuits in the control and relay rooms". The response in the answer key addresses the "EFFECTS" of operating the Transfer Switches.

Reference: OPM section F-9 pages 3 & 4 (refer to the last paragraph on pg. 4)

NRC RESOLUTION:

Comment Accepted. Answer key changed to "to isolate the affected components from their circuits in the control and relay rooms.

FACILITY COMMENT:
ANSWER 3.13

B. The correct response should be as follows:

To defeat RBIC only 2 channels of the 3 can be bypassed following actuation. (Bypassing all 3 channels on a train would cause the system to re-actuate) On the Reactor Trip Isolation all 3 channels per train are bypassed.

Reference: OPM section F-6, pages 39 & 42

NRC RESOLUTION:

Comment Accepted. The facility comment is added as an additional required answer as follows: "Bypassing all 3 channels on a train would cause the system to re-actuate [0.5]."

FACILITY COMMENT:
ANSWER 4.06

Answer key should be expanded to allow additional correct answers based on ATP 1210-1 (Reactor/Turbine Trip), which are as follows:

- Generator field and output breakers open
- Generator stop valves close
- ICS runs main feedwater back
- Turbine bypass valves control OTSG pressure to allow the plant to achieve the post trip windows.

Reference: ATP 1210-1 pages 2,3 & 4

NRC RESOLUTION:

Comment partially accepted. ICS operation of feedwater will be added, but the remaining three items are already part of the present answer and will not be added to the answer.

FACILITY COMMENT:
ANSWER 4.13

Two combinations of correct responses are possible for this question based on the candidates assumptions. Those answers are as follows:

Case 1: Assume that only the "A" OTSG depressurizes

A. Same as answer key
B. FW-V-17B will not close automatically and the ICS will control it's position automatically.
C. FW-V-5B will only close automatically if FW-V-16B closes to <70% open.
D. FW-V-92B will not close automatically.

Reference: OPM section F-3 page 58, figure 16
OPM section F-10 page 3 section III.5 and page 50 & 51 figures 4 and 4A

Case 2: Assume that both OTSGs depressurize

A. Same as answer key
B. FW-V-17B goes closed when OTSG pressure is <600 psig however, ICS does not control position automatically. The HSPS system shuts the valve independently of the ICS.
C. FW-V-5B goes closed when OTSG is <600 psig, however it will not reopen if FW-V-16B gets to the 90 to 95 percent open position. The HSPS acts to remove the air signal to the valve downstream of the ICS signal, causing the valve to remain shut unless the operator bypasses that portion of the HSPS system.
D. Same as answer key

Reference: OPM section F-10 page 3, section III.5 and page 50 & 51 figures 4 & 4A

NOTE the reference quoted in the question is out of date for current plant conditions.

NRC RESOLUTION:

Comment partially accepted. Case 2 not possible due to the design of MSIVs. Therefore Case 1 can apply. The phrase "Closes automatically" will be removed from part b. and points moved to second sentence in answer. Will add the words "when Startup MFRV reaches 70%" to first line in answer c. Will change answer d. to read "remains open [0.8]."

CHANGES MADE BY THE EXAMINER:

Question 4.02.d.

Deleted part d due to insufficient guidance provided to candidates in stem of the question concerning the possibility of using an answer more than once.
Question value dropped to 1.50 and section value dropped to 24.50 and Total for exam dropped to 99.50.

TMI SENIOR REACTOR OPERATOR EXAMINATION
RESOLUTION OF COMMENTS OF EXAMINATION ADMINISTERED ON 12/8/87

FACILITY COMMENT:
ANSWER 5.04

Key answer is incorrect (calculat. nal error made used, 1xE-10 vice 10xE-10)

The correct response should be "C"

P=Po E((SUR)(t))

P=(10xE-10)E((.75 DPM)(2 min.))

P=3.16 E-08

Note: E(x) = Raise 10 to the power x

References: None required calcu ational error in the answer key.

NRC RESPONSE:

Accept facility comment. Answer key modified.

FACILITY COMMENT:
ANSWER 5.10

E. The correct response should be "REMAIN THE SAME," the delta pressure across the tubes stays constant.
(Primary side pressure @2155 psig with secondary pressure @925 psig)

(The delta pressure ALONG the tubes does decrease)

Reference: OPM section B-4, page 13, figure 5

G. The correct response should be "DECREASES," feedwater temperature decreases as power decreases.

Reference: OPM section F-3, page 51, figure 13

NRC RESPONSE:

Accept facility comment. Answer key modified.

FACILITY COMMENT:
ANSWER 6.02

(Note the answer key for questions 6.01 and 6.02 are reversed)

The correct response to the question should be "D", B & C breakers open plus "E" electronic trip.

Reference: OPM section F-1 page 62

NRC RESPONSE:

Accept facility comment. Answer key modified.

FACILITY COMMENT:
ANSWER 6.04

A. The correct answer should be:

NaOH is added to the RCS following an ESAS actuation of 4# RB pressure. The NaOH tank would than be lined up to the suction header of the LPI pumps (DH-P-1A/1B) and the Building Spray pumps (BS-P-1A/1B). From this line up it would than be possible to inject NaOH into the RCS.

The deborating demins are not part of the RCS. NaOH is used to regenerate the demins following exhaustion, while they are isolated from the system.

Reference: OPM section B-6, page 36, figure 9
OPM section B-9, pages 3 & 5
OPM section F-6, page 16

NRC RESPONSE:

Accept facility comment. Answer key modified.

FACILITY COMMENT:
ANSWER 6.06

If the candidate assumed a 1600# ESAS actuation the following valves will reposition and pumps receive a start signal:

-MU-V-2A/2B
-MU-V-14A/14B
-MU-P-1A/1C
-MU-V-36/37
-MU-V-16A/16B/16C/16D
-MU-V-1B

If the candidate assumed a 4# ESAS actuation the following additional valve would reposition:

-MU-V-3

Reference: OPM section F-6 page 44

NRC RESPONSE:

Facility comment is noted. Question will be graded based on applicant's assumption.

FACILITY COMMENT:
ANSWER 6.10

The correct response should be as follows:

Leak rate is more than 4%
1200 GPM; % of signal equals ~35%
with 5% error flow is ~1250 GPM
Flow differential is 1250-1200 GPM = 50 GPM
50/1200 = 4.16% (+ or - based on tolerance band)

Reference: OPM section D-4, page 43

NRC RESPONSE:

Design flow was changed from 2100 gpm to 1200 gpm in the answer key. All other facility values are within the tolerances in the answer key and the key was not changed.

FACILITY COMMENT:
ANSWER 7.04

B. Valves RC-V-2 and/or RC-V-3 are only required to be cycled if they have been closed.

Should accept the above response from the candidate or the response in the answer key for full credit.

Reference: OP 1102-11, page 5, item 10
OPM section B-1 page 36 item 21

NRC RESPONSE:

The question asked for a reason for cycling the spray valves and did not ask when the valves needed to be cycled. The answer key was modified to accept discussion of damage or binding due to thermal transient.

FACILITY COMMENT:
ANSWER 7.05

B. Alternate acceptable response should be as follows:

Throttle EFW flow to stabilize the plant since the EFW throttling criteria are met.

Reference: ATP 1210-10, page 5

NRC RESPONSE:

Accept facilit comment. Answer key modified.

FACILITY COMMENT:
ANSWER 7.06

A. Answer should be "Damage may occur due to movement of the Carriage Upender mechanism (due to system interlocks being deenergized)

Reference: 1507-3, page 6
OPM section M-1, page 24

NRC RESPONSE:

Accept facility comment. Answer key modified.

FACILITY COMMENT:
ANSWER 7.10

D. Alternate acceptable response should be "Operations at power may be continued if no abnormal vibration and increased #2 seal leakage can be handled.

Reference: AP 1203-16, page 3, paragraph 1.2.2

NRC RESPONSE:

The facility's suggested response is contained in the follow up actions in AP 1203-16, which are not required to be known for full credit. The answer key contains the actions required by the immediate actions in AP 1203-16, which are required for full credit. No change made to the answer key.

FACILITY COMMENT:
ANSWER 7.11

Two combinations of correct responses are possible for this question based on the candidates assumptions. Those answers are as follows:

Case 1: Assume that only the "A" OTSG depressurizes

A. Same as answer key
B. FW-V-17B will not close automatically and the ICS will control it's position automatically.
C. FW-V-5B will only close automatically if FW-V-16B closes to <70% open.
D. FW-V-92B will not close automatically.

Reference: OPM section F-3 page 58, figure 16
OPM section F-10 page 3 section III.5 and page 50 & 51 figures 4 & 4A

Case 2: Assume that both OTSGs depressurize

A. Same as answer key
B. FW-V-17B goes closed when OTSG pressure is <600 psig however, ICS does not control position automatically. The HSPS system shuts the valve independently of the ICS.
C. FW-V-5B goes closed when OTSG is <600 psig, however it will not reopen if FW-V-16B gets to the 90 to 95 percent open position. The HSPS acts to remove the air signal to the valve downstream of the ICS signal, causing the valve to remain shut unless the operator bypasses that portion of the HSPS system.
D. Same as answer key

Reference: OPM section F-10 page 3, section III.5 and page 50 & 51 figures 4 & 4A

NOTE the reference quoted in the question is out of date for current plant conditions.

NRC RESPONSE:

Accepted facility comment. Question graded based on candidates' assumptions.

FACILITY COMMENT:
ANSWER 8.01

There are two possible correct answers to the question, they are:

A and C, based on Tech Spec table 3.21.1, action 20

Reference: Tech Spec section 3.21.1, action 20, pages 3-97, 98, 99.

NRC RESOLUTION:

The question was developed from a licensing event report (LER) that was caused by operators initiating a release prior to taking an initial grab sample. No change made to the answer key.

FACILITY COMMENT:
ANSWER 8.09

B. NOT ALLOWED is the correct answer to the question.

Reference: Tech Spec 3.1.3.4.2 (b) on page 3-6 & 6a

NRC RESPONSE:

Accept facility comment. Answer key modified.

FACILITY COMMENT:
ANSWER 8.13

D. Level of emergency declaration should be UNUSUAL EVENT or ALERT based on paragraph 6.1

OR

Declaration of up to SITE AREA emergency based on the SS/ED's judgement per paragraph 4.1.

NOTE: Plant makeup capabilities still exist using MU-P-1A.

Reference: 6410-IMP-1300.01

NRC RESPONSE:

Accept Unusual Event or Alert. The question examines on the candidates' ability to classify an event using the information provided and the facility's procedures. Candidates' classification of events at higher levels based on shift supervisor judgement does not demonstrate an ability to use the procedures.