October 10, 2001

Mr. Robert G. Byram
Senior Vice President and
Chief Nuclear Officer
PPL Susquehanna, LLC
Susquehanna Steam Electric Station
2 North Ninth Street
Allentown, Pennsylvania

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 REACTOR OPERATOR AND SENIOR REACTOR OPERATOR INITIAL EXAMINATION REPORT NOS. 50-387/01-301 AND 50-388/01-301

Dear Mr. Byram:

This report transmits the results of the reactor operator and senior reactor operator licensing examinations conducted by the NRC during the period of August 10-15, 2001. These examinations addressed areas important to public health and safety and were developed and administered using the guidelines of the “Examination Standards for Power Reactors” (NUREG-1021, Revision 8).

Based on the results of the examination, all four applicants passed all portions of the examination. Performance insights observed during the examination process were discussed between Mr. H. Williams of NRC staff and Mr. J. Helsel on August 31 and between R. Conte of NRC staff and W. Hunt and other members of your staff on September 12, 2001. Results of the examinations were given to Mr. Helsel on September 14, 2001.

There were a large number of changes to the written exam, in excess of 5% (8%), after it was administered. Because many of the changes were related to site specific information, this problem indicated poor quality control by the facility. In accordance with NUREG-1021, Section ES-501, item C.2.c., we request that you reply to this letter and provide your perspective on the problem, including why so many changes were necessary, what actions, if any, have been taken or will be taken to improve future initial licensing examinations.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC’s document system (ADAMS). These records which include the final examinations are available in ADAMS (RO/SRO Written-Accession No. ML012690422; RO Operating Section A-Accession No. ML012690559; SRO Operating Section A-Accession No. ML012690608; RO/SRO Operating Section B- Accession No. ML012700010; RO/SRO Operating Section C-Accession No. ML012700057; Facility Post Examination Comments on the Written Exams - Accession No. ML012690520). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Should you have any questions regarding this examination, please contact me at (610) 337-5183, or by E-mail at RJC@NRC.GOV.

Sincerely,

/RA/

Richard J. Conte, Chief
Operational Safety Branch
Division of Reactor Safety

Docket Nos. 50-387, 50-388
License Nos. NPF-14, NPF-22

Enclosure: Initial Examination Report Nos. 50-387; 50-388/2001-301 with Attachment 1

cc w/encl; w/Attachment 1:
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Distribution w/encl; w/Attachment 1:
Region I Docket Room (with concurrences)
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NAME	JWilliams	MShanbaky	RConte		
DATE	09/28/01	10/01/01	10/10/01		

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos:	50-387, 50-388
License Nos:	NPF-14, NPF-22
Report Nos:	50-387/01-301, 50-388/01-301
Licensee:	PP&L Susquehanna, LLC
Facility:	Susquehanna Steam Electric Station Units 1 & 2
Location:	Berwick, PA
Dates:	August 10, 2001 (Written Exam Administration) August 14-15, 2001 (Operating Test Administration) August 16-September 12, 2001 (Grading)
Examiners:	J. Williams, Senior Operations Engineer C. Sisco,Operations Engineer L. Vick, Senior Examiner, NRR
Approved By:	Richard J. Conte, Chief Operational Safety Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000387&388/2001-301; on August 10, 2001 and August 14-15, 2001; Susquehanna Steam Electric Station Units 1 & 2; Initial Operator Licensing Examinations. Four of four applicants (2 ROs, and 2 SRO-upgrades) passed all portions of the examinations.

The written examinations were administered by the facility and the operating tests were administered by three NRC examiners. Because of the relatively large number of post exam changes to the written exam (in excess of 5% (8%)), the quality of the initial submittal was considered problematic.

Report Details

1. REACTOR SAFETY

Mitigating Systems - Reactor Operator (RO) and Senior Reactor Operator (SRO) Initial License Examinations

a. Scope of Review

The NRC examination team reviewed the written and operating examinations and post exam materials submitted by the Susquehanna training staff to verify or ensure, as applicable, the following:

- The examinations were developed in accordance with the guidelines of Revision 8 of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors" and they met the overall quality goals (range of acceptability) of these standards. The review was conducted both in the Region I office and at the Susquehanna training facility. Final resolution of comments and incorporation of test revisions were made during and following the onsite preparation week.

- Simulation facility operation was proper.

- Facility licensee completed a test item analysis on the written exams for feedback into the systems approach to training program.

- Examination security requirements were met.

The NRC examiners administered the operating portion of the exams to all applicants on August 14-15, 2001. Susquehanna training staff administered the written examinations on August 10, 2001.

b. Findings

Grading and Results

All four applicants (2 ROs, and 2 SRO-upgrades) passed all portions of the initial licensing examinations.

Examination Preparation and Quality

The facility had twelve (12) post-examination comments on the written exam and recommended deleting or changing the answers to 8% of the questions on the written exam that it had developed. Five questions were determined to have two correct answers. Comments on four questions did not change the answers as given. Three questions had the correct answer changed. One question was deleted. The NRC disagreed with 3 of the applicant comments. There was a relatively large number of changes to the written exam (in excess of 5% (8%)). The facility initiated Condition Report No. 350472 to investigate the problem. (See ADAMS-Accession No. ML012690520 for the facility post exam comments and for the NRC resolution of the comments.)

Examination Administration and Performance

No findings of significance were identified.

4OA6 Exit Meeting Summary

On August 31 and September 12, 2001 the NRC provided observations associated with the exam to PP&L management. Examination results were provided to the facility on September 14, 2001. License numbers were also provided during the September 14th telephone call.

The NRC expressed appreciation for the cooperation and assistance that was provided during the preparation and administration of the exams by the licensee's training and operation staffs.

KEY POINTS OF CONTACT

Susquehanna Steam Electric Station

M. Trump	Operations Training Supervisor
J. Helsel	Supervisor Nuclear Training
R. Chin	Operations Instructor
J. Seek	Operations Instructor
E. Bowles	SSES Contractor
P. Ballard	SSES Contractor
W. Hunt	Manager-Nuclear Training

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

ITEM NUMBER	TYPE	DESCRIPTION
NONE		

ATTACHMENT 1

SUMMARY OF FACILITY COMMENTS ON THE WRITTEN EXAMS AND NRC RESOLUTION OF THE COMMENTS

1. Question RO 15/SRO 13 (28)

Which one of the following is the bases for a reactor scram on a main turbine trip above 30% reactor power?

A. Provides a backup to the RPV pressure and APRM high scrams.

B. Ensures RPV water level remains above the dryer separator skirt.

C. Protects the reactor from the pressure effects of a loss of heat sink.

D. Anticipates a positive reactivity addition from a loss of feedwater heating.

Answer: C (No Change)

Facility Comment Summary: An applicant requested the bases for the answer be re-evaluated. The question was reviewed by the facility and no changes made.

NRC Resolution: The applicant did not provide an alternate answer. However, the NRC also evaluated the answer and concluded that Technical Specification Bases B.3.3.1.1.8 supports the answer given. The facility conclusion is correct.

2. Question RO 23/SRO 21 (33)

Unit 2 is shutdown with the following:

- "2A" RBCCW and "2A" TBCCW are aligned to ESW.
- Loop "A" of ESW is isolated from the Diesel Generators (DGs)

A loss of off-site power occurs

- DG output breaker 1A20404 fails to close
- "B" ESW Pump fails to start

Assuming NO operator actions, which one of the following is required?

A. Trip ALL the DGs in four and one half (4.5) minutes.

B. Trip DG "B" and DG "D" in four and one half (4.5) minutes.

C. Trip DGs "A", "B" and "C" in four and one half (4.5) minutes and DG "D" in eight (8) minutes.

D. Trip DGs "A" and "C" in four and one half (4.5) minutes and DGs "B" and "D" in eight (8) minutes.

Answer: C (changed to A.   B is also correct)

Facility Comment Summary: All diesel generators (DGs) are running loaded without cooling water. Change the answer from “C” to “A”.

NRC Resolution: Based upon the information provided, accept the answer change from “C” to “A” because all DGs are running loaded without cooling water. Answer ‘B’ (2 of the 4 DGs) is a subset of ‘A’ (4 of 4 DGs) and therefore is correct also.

3. Question RO 33/SRO 29 (36)

Plant conditions are as follows:

- Reactor has been in Cold Shutdown for 2 days following power operation.
- Reactor water level is +87 inches.
- Both reactor recirc pumps are tagged out of service.
- Shutdown cooling has isolated and the shutdown cooling suction valves cannot be opened.

Which one of the following operator actions will reverse or prevent vessel stratification AND provide alternate decay heat removal?

A. Place Reactor Water Cleanup in service in recirculation.

B. Insert a manual scram to maximize Control Rod Drive flow to the RPV.

C. Start a second Control Rod Drive pump and maximize cooling water D/P.

D. Begin rejecting water with Reactor Water Cleanup while injecting with CRD.

Answer: A   (D is also correct)

Facility Comment Summary; Accept answers “A” and “D” as correct. The reference material provided supports both answers as acceptable methods.

NRC Resolution: The facility comment is accepted; methods described in choices “A” and “D” are supported by procedure ON-149-001, discussion (page 24 of 31) and Attachment B (pages 29-30 of 31).

4. Question RO 50

Unit 1 is operating at 65% power while a surveillance test is being performed on the recirculation drive flow instruments. During the surveillance the Mode Switch for the A Flow Unit is placed in zero (0) without first bypassing the flow unit.
Which one of the following will occur and what action is required?

A. Several control room annunciators alarm and a rod block occurs, NO half scrams occur, bypass the A Flow Unit.

B. Several control room annunciators alarm and a full scram occurs, enter ON-100-101, SCRAM and take the immediate actions.

C. Several control room annunciators alarm and a rod block and half scram occur, bypass the A Flow Unit and reset the half scram.

D. Control room annunciator APRM/RBM FLOW REFERENCE OFF NORMAL activates, NO rod block or trips occur, bypass the A FLOW UNIT.

Answer: D (Change to A)

Facility Comment Summary; Change the correct answer from “D” to “A”. Per the reference material provided, the effect of the action is a (1) Rod Out Block and (2) APRM/RBM Flow reference off normal.

NRC Resolution: Accept the answer change from “D” to “A”. Per the reference material provided (alarm response procedures and logic diagrams) the effect of placing the mode switch for the ‘A’ Flow Unit out of operate will result in a rod block as well as control room annunciator alarms.

5. Question RO 52/SRO 41 (54)

Unit 1 was at 30% power when a reactor scram occurred on a loss of vacuum after circulating water was lost. After the initial scram actions were taken the following occurred:

- Reactor Core Isolation Cooling (RCIC) was placed in pressure control mode per OP-150-001.
- Workers in the Reactor Building bump instrument Rack 1C004 causing a Division 1 Low RPV Level Trip (-30 inches).

Which one of the following is the effect on RCIC and the reasons for that effect?

A. No effects because RCIC will NOT realign after being manually placed in this line-up.

B. No effects, RCIC remains in pressure control mode, because only one division is effected.

C. RCIC automatically aligns for RPV injection because only one division is required for system initiation.

D. RCIC automatically aligns for RPV injection after RPV level lowers to actuate in Division 2 RPV Level Trip.

Answer: C

Facility Comment Summary: An applicant stated that the stem does not state that both channels have tripped on the rack. If only one switch actuated, “B” could be correct also. The facility did not accept the comment because the stem stated “Division 1".

NRC Resolution: Agree with the facility and not accept “B” as correct since there is no basis for making the assumption that only one switch actuated.

6. Question RO 54/ SRO 43 (56)

With Unit 1 at 85% power when a load reject and loss of off-site power occur. The diesel generators start and power their associated buses.

Which one of the following describes the effect on Drywell Cooling?

The operating drywell unit coolers trip, then...

A. remain shutdown until manually started.

B. restart when the diesels start and are cooled by RBCW.

C. restart when the diesels start and are cooled by RBCCW.

D. restart when the diesels start and run without cooling water.

Answer: D (Change to C)

Facility Comment Summary: Change the answer from “D” to “C”. On a loss of off-site power (LOOP), Reactor Building Chilled Water (RBCW) Drywell loads shift to Reactor Building Closed Cooling Water (RBCCW).

NRC Resolution: Comment accepted, based upon procedure ON-104-001, “Unit 1 Response to Loss of All Offsite Power”, Step 3.11, drywell cooling shifts to RBCCW when RBCW trips on the LOOP.

7. Question RO 55/SRO 44 (57)

Unit 1 has scrammed and the MSIVs isolated. The cause of the isolation has been corrected and the MSIV isolation logic reset. With RPV pressure greater than 600 psig which one of the following is required to re-open the MSIVs?

A. Drain the steam lines, bypass the MSIVs with the steam drains, lower the D/P to less than 200 psid the open the inboard then the outboard MSIVs.

B. Open the inboard MSIVs, drain the steam lines, bypass the outboard valves with the steam drains, lower the D/P to less than 50 psid then open the outboard MSIVs.

C. Drain the steam lines, open the outboard MSIVs, bypass the inboard valves with the steam drains, lower the D/P to less than 200 psid then open the inboard MSIVs.

D. Open the outboard MSIVs, drain the steam lines, bypass the inboard valves with the steam drains, lower the D/P to less than 50 psid then open the inboard MSIVs.

Answer: D (Accept “B” also)

Facility Comment Summary: Distractor “B” could also be correct because ON-184-001, “Main Steam Line Isolation Quick Recovery” has the inboard MSIVs opened first. OP-184-001, “Main Steam System” directs opening the outboard MSIVs first.

NRC Resolution: Either the inboard or outboard MSIVs can be opened first, depending upon the procedure used. Answers “B” and “D” are correct per the referenced procedures. Facility comment is accepted.

8. Question RO 66/SRO 51 (68) **Answer Key was changed before the exam. The Chief Examiner was notified of the change before the exam by the facility and the Chief Examiner indicated a review would be conducted after the exam.**

Control rods are being withdrawn during a plant startup. The following conditions exist:

- The Rod Worth Minimizer (RWM) is in operation
- Control rods are being withdrawn in group 4
- Only one control rod remains to be withdrawn in group 4
- The operator attempts to select and withdraw a control rod in group 5

Which one of the following describes the response of the control rod in group 5, the response of the RWM, and the required action?

The control rod...

A. will NOT withdraw, a control rod withdraw block will be applied to this rod. Select the correct control rod in group 4.

B. will NOT withdraw, a select block will be applied to the control rod in group 5. Bypass the RWM then select the correct control rod in group 4.

C. will withdraw to its withdraw limit, the last rod in group 4 will be identified as an insert error. Promptly insert the control rod in group 5 to position 00.

D. will withdraw only one notch, then control rod withdrawal blocks will be applied to all other control rods. Position the control rod in group 5 to its intended position.

Answer: D (Changed answer to "C" before the exam)

Facility Comment Summary: Before the exam was administered the facility determined that the correct answer was "C" rather than "D" and changed the answer to "C".

NRC Resolution: Do not accept the facility comment. The facility provided reference material that did not support any answer as correct. The question was deleted.

Based on the information provided, the RWM is designed to allow one or two rods in the selected group to remain at positions lower than their withdrawal limits and the next higher group can be latched and startup allowed to proceed. The question stem asks for the required action and the proposed answer states "promptly insert the control rod in group 5 to position 00". This action is not required. The stem does not state whether the selection of the group 5 control rod and leaving the last group 4 control rod inserted was an error. Therefore the action is not required. The question has no correct answer.

9. Question RO 67

Unit 1 is operating at 80% power with 76 Mlbm/hr core flow when a spurious feedwater flow signal causes a recirculation flow control runback. After the runback the following conditions exist:

- APRMs oscillating between 44% and 48% power
- Core flow is 42 Mlbm/hr
- Green lights are illuminated above RX RECIRC LIMITER 1 RUNBK RESET pushbutton
- One (1) center region C-level LPRM upscale alarm is sealed in
- Two (2) peripheral A-Level LPRM downscale alarms are sealed in

In accordance with ON-164-002, Recirc Drive Flow Instrument Failure, and the Power/Flow Map, which one of the following actions is required?

A. Raise core flow to at least 44 Mlbm/hr.

B. Place the reactor mode switch in SHUTDOWN.

C. Monitor for power instabilities and wait for RE instructions.

D. Insert control rods in accordance with the cram array to less than 40% power.

Answer: A (Accept "D" also)

Facility Comment Summary: Accept answers "A" and "D" as correct. Procedure On-164-002, "Loss of Reactor Recirculation Flow," directs actions specified on the Power/Flow Map. The Power/Flow Map directs actions in ON-178-002, "Core Flux Oscillations". ON-178-002, Step 3.4.3 allows insertion of control rods or raising core flow.

NRC Resolution: Agree with facility comment and accept two answers based upon the references provided. The Power/Flow Map directs actions in ON-178-002, "Core Flux Oscillations", which allows control rod insertion or raising core flow. Based upon the given reactor power and flow conditions, raising core flow to at least 44 Mlbm/hr or inserting control rods to obtain a power level of less than 40% will result in exiting Region II (an unstable region) on the Power/Flow Map.

10. Question RO 69/SRO 53 (70)

A shutdown and cool down is in progress on Unit 1. Per OP-149-002, RHR Shutdown Cooling, the required level band is established then Shutdown Cooling (SDC) is placed into service with one Reactor Recirculation (RR) loop shutdown.

One (1) hour after establishing SDC, the operating RR pump trips. Which one of the following described the action to be taken for reactor water level?

Adjust reactor water level...

A. to a new band of +35 to +50 inches.

B. to a new band of +90 to +100 inches.

C. maintaining level within the band established before the event.

D. maintaining level above the band established before the event.

Answer: C

Facility Comment Summary: An applicant stated that the new band of +90 to +100 inches was also the established band prior to the event. Therefore accept answers "B" and "C". The facility noted that this was not a new band and did not accept the comment. The facility recommended changing distractor "B" to +60 to +90 inches to avoid confusion.

NRC Resolution: Agree with the facility that the level band was not a new band and do not accept answer "B". The water level band of +90 to +100 inches was in effect before the event and therefore distractor 'B' is incorrect because it is not a new band. NRC agrees that distractor "B" was poorly written and should be changed before installation for bank use.

11. Question RO 92/SRO 68 (89)

During a reactor heat up the following temperature readings are recorded on Attachment A of SO-100-011, Reactor Vessel Temperature and Pressure Recording:

- 0800 - 242 degrees F
- 0815 - 263 degrees F
- 0817 - Startup was temporarily halted
- 0830 - 239 degrees F
- 0845 - 268 degrees F
- 0900 - 311 degrees F

Per GO-100-002, which one of the following is the maximum allowable temperature at 0915?

A. 329 degrees F

B. 339 degrees F

C. 353 degrees F

D. 363 degrees F

Answer: A

Facility Comment Summary: A caution in SO-100-011 warns the operator to limit the heat up rate to 25 degrees F per 15 minutes. The question asks for a 45 minute change, so the limit would be 75 degrees F heat up and the correct answer would be 314 degrees F, which is less than the lowest temperature given as a response. Delete the question.

NRC Resolution: Comment not accepted.

The question asks for the maximum allowable temperature at 0915. The change in temperature is measured in 15 minutes periods and the 25 degree temperature change guideline was already violated for the last two 15 minute periods. Keeping the temperature change to 25 degrees or less per 15 minutes is to prevent violating the technical specification limit of 100 degrees F per hour. The facility has an administrative limit of 90 degrees F per hour, which gives 329 degrees F at 0915 as the maximum allowable temperature to stay at the administrative limit of 90 degrees F for the hour. The last 15 minutes would have no temperature rise.

"A" is the correct answer. No temperature change is allowed in the last 15 minutes of the hour.

12. Question RO 95/SRO 70 (93)

Unit 1 is operating at 95% power when the High Pressure Coolant Injection (HPCI) system initiates on a spurious high drywell pressure signal. Which one of the following sets of parameters result from this transient?

	APRM Power	Total Core Flow	Generator Mwe	Feedwater Flow
A.	RISE	NO CHANGE	LOWER	LOWER
B.	NO CHANGE	LOWER	RISE	NO CHANGE
C.	RISE	NO CHANGE	RISE	LOWER
D.	NO CHANGE	LOWER	LOWER	NO CHANGE

Answer: C (Accept “A” also)

Facility Comment Summary: Accept “A” or “C” as correct since the stem does not specify steady state conditions for the resulting parameters. A small fraction of the steam flow is diverted from the turbine/generator to feed the HPCI turbine and this results initially in a lowering of generator Mwe as shown on the simulator traces.

NRC Resolution: Accept answers “A” and “C” as correct. The question was written to solicit an answer based upon steady state conditions after the transient. However, this was not specified in the stem and before steady state conditions are established, the conditions described in distractor ‘A’ would exist for a short time.