November 26, 2008

Mr. Keith J. Polson, Vice President
Nine Mile Point Nuclear Station, LLC
P.O. Box 63
Lycoming, NY 13093-0063

SUBJECT: NINE MILE POINT NUCLEAR STATION, UNIT 1 - NRC EXAMINATION REPORT 05000220/2008302

Dear Mr. Polson:

On October 13, 2008 the U.S. Nuclear Regulatory Commission (NRC) completed an examination at Nine Mile Point Nuclear Station, Unit 1. The enclosed report documents the examination findings, which were discussed on November 14, 2008, with Mr. Robert Brown and other members of your staff.

The examination included the evaluation of three applicants for reactor operator licenses, five applicants for instant senior operator licenses and one applicant for an upgrade senior operator license. The written and operating examinations were developed using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1. The license examiners determined that four of the nine applicants satisfied the requirements of 10 CFR Part 55, and the appropriate licenses have been issued. The issuance of two of these four licenses is on hold pending completion of deferred eligibility requirements.

No findings of significance were identified during this examination.

In accordance with 10 CFR 2.390 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). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Samuel L. Hansell, Jr., Chief
Operations Branch
Division of Reactor Safety

Enclosure: NRC Examination Report 05000220/2008302

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Division of Reactor Safety

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EXAMINATION REPORT
U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Dockets:	50-220
Licenses:	DPR-63
Report :	05000220/2008302
Licensee:	Constellation Energy Group
Facility:	Nine Mile Point Nuclear Station, Unit 1
Location:	Oswego, NY
Dates:	October 6 to October 23, 2008
Inspectors:	Joseph M. D'Antonio, Chief Examiner, Operations Branch Gil Johnson, Operations Engineer Brian Fuller, Operations Engineer
Approved By:	Samuel L. Hansell, Jr., Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000220/2008302; October 6-23, 2008; Nine Mile Point Nuclear Station, Unit 1; Initial Operator Licensing Examination Report.

NRC examiners evaluated the competency of three applicants for reactor operator licenses, five applicants for instant senior operator licenses and one applicant for an upgrade senior operator license at Nine Mile Point Nuclear Station, Unit 1. The facility licensee developed the examinations using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1. The written examination was administered by the facility on October 13, 2008. Three NRC examiners administered the operating tests on October 6-10, 2008. The license examiners determined that four of the nine applicants satisfied the requirements of 10 CFR Part 55, and the appropriate licenses have been issued.

A. NRC-Identified and Self-Revealing Findings

No findings of significance were identified.

B. Licensee-Identified Violations

None.

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other Activities (Initial Operator License Examination)

.1 License Applications

a. Scope

The examiners reviewed all nine license applications submitted by the licensee to ensure the applications reflected that each applicant satisfied relevant eligibility requirements. The applications were submitted on NRC Form 398, "Personal Qualification Statement," and NRC Form 396, "Certification of Medical Examination by Facility Licensee." The examiner also audited two of the license applications in detail to confirm that they accurately reflected the subject applicant's qualifications. This audit focused on the applicant's experience and on-the-job training, including control manipulations that provided significant reactivity changes.

b. Findings

No findings of significance were identified.

.2 Operator Knowledge and Performance

a. Examination Scope

On October 13, 2008, the licensee proctored the administration of the written examinations to all nine applicants. The licensee staff graded the written examinations, analyzed the results, and presented their analysis to the NRC on October 23, 2008. The NRC examination team independently graded the written examinations. The NRC results were consistent with the licensee grades.

The NRC examination team administered the various portions of the operating examination to all nine applicants on October 6-10, 2008. The three applicants for reactor operator licenses participated in two dynamic simulator scenarios, in a control room and facilities walkthrough test consisting of eleven system tasks, and an administrative test consisting of four administrative tasks. The five applicants seeking an instant senior operator license participated in three dynamic simulator scenarios, a control room and facilities walkthrough test consisting of ten system tasks, and an administrative test consisting of five administrative tasks. The one applicant for an upgrade senior operator license participated in three dynamic simulator scenarios, a control room and facilities walkthrough test consisting of five system tasks, and an administrative test consisting of five administrative tasks.

b. Findings

Four of the applicants passed all parts of the operating test. One reactor operator applicant failed the written examination. One reactor operator applicant, one senior reactor operator upgrade applicant, and two senior reactor operator instant applicants failed the simulator portion of the operating test. For the written examinations, the reactor operator applicants' average score was 83.09 percent and ranged from 79.45 to 89.04 percent, the senior operator applicants' average score was 88.27 percent and ranged from 80.4 to 92.7 percent. The overall written examination average was 86.54 percent. The text of the examination questions, the licensee's examination analysis, and the licensee's post-examination comments may be accessed in the ADAMS system under the accession numbers noted in the attachment.

Chapter ES-403 and Form ES-403-1 of NUREG 1021 require the licensee to analyze the validity of any written examination questions that were missed by half or more of the applicants. The licensee conducted this performance analysis for eight questions that met these criteria and submitted the analysis to the chief examiner.

On October 23, 2008, the licensee submitted post-examination comments for 6 questions. The licensee recommended deleting two questions, changing the correct answer for two questions, and accepting two answers for two questions. After reviewing the licensee's comments, the NRC decided to delete three questions, accept two responses for one question, change the answer for one question, and make no change to the remaining question. See Attachment 2 for the licensee comments and Attachment 3 for NRC responses.

.3 <u>Initial Licensing Examination Development</u>

a. <u>Examination Scope</u>

The facility licensee developed the examinations in accordance with NUREG-1021, Revision 9, Supplement 1. All licensee facility training and operations staff involved in examination preparation and validation were on a security agreement. The facility licensee submitted both the written and operating examination outlines on July 24, 2008. The chief examiner reviewed the outlines against the requirements of NUREG-1021, Revision 9, Supplement 1, and provided comments to the licensee. The facility licensee submitted the draft examination package on August 14, 2008. The chief examiner reviewed the draft examination package against the requirements of NUREG-1021, Revision 9, Supplement 1, and provided comments to the licensee on the examination on August 21, 2008. The NRC conducted an onsite validation of the operating examinations and provided further comments during the week of September 15, 2008. The licensee satisfactorily completed comment resolution on September 29, 2008.

b. <u>Findings</u>

The NRC approved the initial examination outline and advised the licensee to proceed with the operating examination development.

The examiners determined that the written and operating examinations initially submitted by the licensee were within the range of acceptability expected for a proposed examination.

No findings of significance were identified.

.4 <u>Simulation Facility Performance</u>

a. Examination Scope

The examiners observed simulator performance with regard to plant fidelity during the examination validation and administration.

b. Findings

No findings of significance were identified.

.5 Examination Security

a. Examination Scope

The examiners reviewed examination security for examination development and during both the onsite preparation week and examination administration week for compliance with NUREG-1021 requirements. Plans for simulator security and applicant control were reviewed and discussed with licensee personnel.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

The chief examiner presented the examination results to Mr. Robert Brown, General Supervisor of Operations Training and other members of the licensee's management staff on November 14, 2008.

The licensee did not identify any information or materials used during the examination as proprietary.

ATTACHMENT 1: SUPPLEMENTAL INFORMATION
ATTACHMENT 2: FACILITY POST EXAM COMMENTS
ATTACHMENT 3: NRC RESOLUTION OF FACILITY POST EXAM COMMENTS

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

Tom Shortell, Operations Training Manager
Robert Brown, General Supervisor, Operations Training

NRC Personnel

Joseph D'Antonio, Senior Operations Engineer

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

NONE

Opened and Closed

NONE

Closed

NONE

Discussed

NONE

ADAMS DOCUMENTS REFERENCED

Accession No. ML083230440 – FINAL-Written Exam
Accession No. ML083230370 – FINAL-Operating Exam Section A
Accession No. ML083230376 – FINAL-Operating Exam Section B
Accession No. ML083230387 – FINAL-Operating Exam Section C

ATTACHMENT 2

FACILITY POST EXAM COMMENTS

In accordance with Nine Mile Point Administrative Procedure NMP-TR-1.01-21, Initial License Training Exam Writer's Guide, and NUREG 1021, Revision 9, Supplement 1, Sections ES-402 and ES-501, an examination analysis was conducted on the 2008 Unit 1 Initial NRC Written Examination. This analysis included a detailed review of each question missed by 50% or more of the applicants. Three questions were identified during this analysis as requiring a change in grading. The table below summarizes the findings of this detailed question review.

Unit 1 2008 NRC Exam Analysis - High Miss Questions

Question	Percent Correct	New, Modified, Bank	Higher or Lower Cognitive	System or Topic	Justification
12	22%	B	H	FW, AC	Recommend changing keyed answer from D to C. Research indicates that Feedwater Booster pumps 11 and 13 will not restart following slow transfer of their Powerboards. Therefore Feedwater pump 11 will not start on a low level HPCI signal due to an interlock requiring either Feedwater Booster pump 11 or 13 to be running.
15	33%	N	H	RBCLC, SOP	Recommend accepting both B and C as correct answers. Stem conditions directly support C as correct based on the SOP guidance. Additionally, the stem conditions present the candidates with a situation similar to what they have seen in a training simulator scenario. Given this training and a bias for conservative decision making, the candidates believed that the RBCLC pumps were no longer performing their design function. Therefore execution of the override for no RBCLC pumps running (ie not performing design function) is acceptable, as in choice B.
33	33%	N	L	OG, EOP	Six candidates did not know the method of using Offgas as an alternate pressure control system in EOP-1 attachment 21. This knowledge deficiency was discussed during the exam review. The EOP attachment directs only the use of the valve controller in manual from the control room.
49	33%	N	L	SOP	Six candidates chose B, which is incorrect based on SOP guidance. Control Room Evacuation procedure directs use of Condensate/FW AND CRD. Choice B excludes use of CRD. The SOP was reviewed during exam review and the candidates agreed CRD was a viable system for use in this situation.

51	44%	N	L	CREVS	Five candidates did not remember operation/interlocking of CREVS and smoke purge system. This knowledge deficiency was discussed during the exam review.
69	44%	N	L	TS	Five candidates did not recognize that in cold shutdown primary containment is not required. The Tech Spec requirements were discussed during exam review.
79	50%	N	H	CRD, EOP	Three candidates did not recognize indications of malfunctioning CRD FCV. During the exam review, the stem conditions were discussed to ensure candidates could diagnose the malfunctioning FCV based on the indications given.
86 (SRO 11)	0%	N	H	FW, IA, SOP	Recommend changing keyed answer from C to B. All six candidates eliminated C as a possible answer, since taking manual control of hotwell level will not prevent the loss of HPCI unless the Feedwater flow control valves are also manually controlled. Therefore the actions in C would not prevent the loss of HPCI. All six candidates chose B based on degraded instrument air conditions. With indication of clogging in the IA filters and no way of bypassing these filters, the candidates determined that the best course of action with the given choices was to bypass the upstream IA dryers to eliminate as much flow restriction as possible.

Additionally, an examination review session was held to solicit applicant feedback on each question. This examination review session identified an additional three questions requiring a change in grading. The table below summarizes the three additional questions requiring a change in grading.

Question	Percent Correct	New, Modified, Bank	Higher or Lower Cognitive	System or Topic	Justification
2	56%	N	H	EOP	Choices B, C, and D are incorrect as stated in the question justification. Choice A is also incorrect. The question gives conditions of all control rods inserted, with current conditions satisfying the Heat Capacity Temperature Limit, with no indication of degraded Containment Spray system availability. Following RPV Blowdown, the heat addition to the containment would be well within the capacity of the Torus Cooling mode of Containment Spray. Therefore containment venting due to approaching the Primary Containment Pressure Limit is not a reasonable answer.

29	56%	N	H	Fire	Additional technical information has been found that proves that choice D is not true for all fire suppression. Detection circuits exist that need multiple disconnect switches repositioned to defeat alarms and suppression. This supports choice A as correct due to stem wording of a singular disconnect switch. Wet pipe suppression systems also exist in which the zone disconnect switch will defeat alarms, but not suppression. This supports choice C as correct.
67	78%	N	H	RMCS, SOP	Recommend accepting both C and D as correct answers. In the situation provided, SOP-1, Reactor Scram, directs inserting control rods using OP-5. OP-5 section H.23 directs use of the Control Rod Movement switch for control rod insertion. This supports the originally proposed answer of D. Precaution and Limitation 9 of OP-5 provides additional guidance on use of the Emergency In switch. Given the degraded situation of multiple control rods not fully inserting following a reactor scram, it is reasonable to believe that use of the Emergency In switch would be authorized for control rod insertion. This bypasses the RMCS timer, allowing control rod insertion without waiting for timer reset.

In summary, we propose the following questions be DELETED from the examination:

Question	Basis	Number of Affected Applicants
2	No correct answer	All
29	Multiple correct answers	All

We propose the following questions have TWO CORRECT ANSWERS:

Question	Basis	Number of Affected Applicants
15	Unclear stem conditions combined with additional procedural guidance	2 of 3 ROs 4 of 6 SROs
67	Additional procedural guidance discovered	1 of 3 ROs 1 of 6 SROs

We propose the following question has ONLY ONE CORRECT ANSWER, but it is NOT THE ORIGINALLY APPROVED ANSWER:

Question	Basis	Number of Affected Applicants
12	Technical error	1 of 3 ROs 5 of 6 SROs
86	Stem focus	All SROs

In accordance with ES-402, Section E.5, the following pages provide the analyses, justification for change, and our recommended disposition for each of these questions.

We present each of these questions in the following format:

Part 1 - The original question as approved by the NRC and administered to the applicants.

Part 2 - The justification for change, including an Analysis, Conclusions, and the Recommendation.

Part 3 - A copy of the reference documentation that supports the recommended change.

Question 2
Part 1 - Original Question

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	239002 K1.07	
	Importance Rating	3.6	

Knowledge of the physical connections and/or cause- effect relationships between RELIEF/SAFETY VALVES and the following: Suppression pool

Proposed Question: Common 2

Following a reactor scram, the following conditions exist:

- All controls rods have inserted
- ERV 113 has failed OPEN
- Torus temperature is 146°F and rising
- Torus water level is 11.0 feet and slowly rising
- RPV pressure is 500 psig and slowly lowering

The SRO has directed an RPV Blowdown.

What is the consequence of opening additional ERVs at this time?

A. Containment venting may be required to remain within the Primary Containment Pressure Limit

B. Torus water temperature will exceed the Containment Spray NPSH Limit for minimal flows even with Containment venting

C. Torus water temperature will exceed its design limit before Reactor pressure reaches the Minimum RPV Flooding Pressure

D. Primary Containment pressure will exceed its design limit before Reactor pressure reaches the Minimum RPV Flooding Pressure

Proposed Answer: A

Explanation (Optional):

A. Correct – The purpose of blowing down when exceeding HCTL is to lower RPV pressure to the point where the rate of energy transfer from the RPV to the primary containment with 3 ERVs open is within the capacity of the containment vent. This strategy leads to the possibility of further venting to control containment pressure.

B. Incorrect - As containment pressure rises it will raise NPSH "*Overpressure = Torus Pressure + 0.433 (Torus Water Level – 4.5)*" therefore NPSH is not a concern.

C. Incorrect - Although Torus water temperature may be exceeded the Torus water temperature design limit will occur after Reactor pressure reaches the Minimum RPV Flooding Pressure.

D. Incorrect - Although Primary Containment pressure may be exceeded it would be after Reactor pressure reaches the Minimum RPV Flooding Pressure.

Technical Reference(s): EOP-SAP Bases, Sect 1.2.E (Attach if not previously provided)

Proposed references to be provided to applicants during examination: N1-EOP-4 or at a minimum EOP Curves.

Learning Objective: N1-218000-RBO-12 (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content:
55.41 X
55.43 ______

Question 2
Part 2 - Justification

Analysis:

The stem conditions present a successful reactor scram with a stuck open ERV and an RPV Blowdown required. The initial conditions indicate that the Heat Capacity Temperature Limit (HCTL) is satisfied. The question asks what the consequences would be of opening additional ERVs at this time.

The basis for HCTL is presented in Part 3. The basis states "the Heat Capacity Temperature Limit ensures that there is sufficient heat capacity in the torus to sustain a blowdown".

The stem conditions do not present any information that would put into question the availability of the Containment Spray system to either cool the torus in the Torus Cooling mode or spray the Containment in the Containment Spray mode. The stem conditions do not present any information that would put into question the ability to maintain RPV water level and adequate core cooling.

This scenario of energy addition to the containment is bounded by the design basis loss of coolant accident analysis. An excerpt from this analysis is presented in Part 3. In this worst case scenario, peak Drywell pressure reaches 35 psig and equalizes rapidly to 22 psig. These peak pressures alone are not in excess of that allowed by the Primary Containment Pressure Limit (PCPL). The PCPL curve is presented in Part 3. Given that this rapid, limiting blowdown of energy from the RPV to the Containment does not result in exceeding PCPL and the availability of the Containment Spray system to provide heat removal from the Containment, it is not reasonable to believe that venting of the Containment would be required in the scenario presented by this question.

Conclusions:

The originally accepted answer is not technically valid. Therefore no correct answer exists for this question.

Recommendation:

On the basis of there being no correct answer to this question, delete question 2 from the examination.

Question 2
Part 3 - Reference Documentation

A. NER-1M-095, NMP1 Emergency Operating Procedures and Severe Accident Procedures (EOP/SAP) Basis Document, Section 3.6, Heat Capacity Temperature Limit

B. UFSAR Section VI B.1.2, Design Basis Accident (DBA), and B.1.3, Containment Heat Removal

C. Primary Containment Pressure Limit curve

Question 12
Part 1 - Original Question

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000, K6.03	
	Importance Rating	2.9	

(K&A Statement) Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE COOLANT INJECTION SYSTEM : AC Power

Proposed Question: Common 12

The plant is operating at 50% power with the following conditions:

- Feedwater pump 12 is in PTL with a red clearance tag attached
- A fault in the T10 load tap changer causes T10 output to fall below 3,200 VAC

Without operator action, which one of the following describes the plant response to this event?

A. PB 11 and PB 12 fast transfer. Feedwater Pump 11 does NOT start to control Reactor water level.

B. PB 11 and PB 12 fast transfer. Feedwater Pump 11 starts to control Reactor water level.

C. PB 11 and PB 12 slow transfer. Feedwater Pump 11 does NOT start to control Reactor water level.

D. PB 11 and PB 12 slow transfer. Feedwater Pump 11 starts to control Reactor water level.

Proposed Answer: D

Explanation (Optional):

D. Correct - The 24 kV Main Generator Bus Duct is tapped to provide power to Normal House Service Transformer 10. This transformer supplies two separate busses: Power Board 11 and Power Board 12. Low voltage initiation of transfer (slow transfer) is delayed to ensure override of transient disturbances. In this case special relays delay closure of the Reserve Breaker until the bus voltage has decayed to 20% of normal. This delay leads to a Rx scram and subsequent Turbine trip. HPCI is auto-initiated by the Turbine trip, as well as low RPV water level. Feedwater pump 11 starts on the HPCI initiation to control RPV water level.

A. Incorrect - Under degraded voltage conditions the 27TR relays in the breaker control logic prevent the rapid transfer to reserve power and force a delay of at least 27 cycles (about1/2 second) before the auto-closing function is enabled. Feedwater pump 11 will start on HPCI initiation following a Rx scram and Turbine trip.

B. Incorrect - Under degraded voltage conditions the 27TR relays in the breaker control logic prevent the rapid transfer to reserve power and force a delay of at least 27 cycles (about1/2 second) before the auto-closing function is enabled.

C. Incorrect - Feedwater pump 11 will start on HPCI initiation following a Rx scram and Turbine trip.

Technical Reference(s): N1-OP-30, Sect. B. (Attach if not previously provided)
N1-OP-16, Sect. B

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262000-RBO-11 (As available)

Question Source: Bank # N1-262000-RBO-11-Q-01
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 X

Question 12
Part 2 - Justification

Analysis:

The stem conditions present a condition where Powerboard 11 and 12 are experiencing a degraded voltage. This leads to a slow transfer of the power supply breakers, in which the normal supply breaker from the station service transformer opens, and the alternate supply breaker from the reserve transformer closes after voltage has decayed on the Powerboard.

The delayed transfer of the Powerboard results in a loss of multiple loads, including both Circulating Water pumps, four of the five Reactor Recirculation pumps, and the Feedwater and Condensate pumps that are part of the High Pressure Coolant Injection (HPCI) system. Prints are included in Part 3 detailing the loads on Powerboards 11 and 12.

The reactor will scram on this transient. Given the initial power level of 100%, RPV water level will shrink below the HPCI initiation setpoint. Under normal circumstances, the HPCI initiation signal would result in Feedwater pump 11 automatically starting.

However, the Feedwater pump 11 starting logic requires either Feedwater Booster pump 11 or 13 to be running. Feedwater Booster pump 11 is powered from Powerboard 11 and would stop operating during the slow transfer. Feedwater Booster pump 13 is powered from Powerboard 12 and would stop operating during the slow transfer. The loss of these two Feedwater Booster pumps would result in autostart of Feedwater Booster pump 12 due to low header pressure. The logic for Feedwater Booster pump 12 is included in Part 3. Feedwater Booster pump 12 is powered from Powerboard 101, which is normally energized from the reserve transformer, and is thus unaffected by the degraded voltage presented in the stem conditions. A print is included in Part 3 detailing the loads on Powerboard 101.

With Feedwater Booster pump 12 supplying header pressure, there is no automatic start signal for Feedwater Booster pumps 11 or 13 once they regain power. Therefore neither Feedwater Booster pump 11 or 13 will be running following the slow transfer. This will prevent Feedwater pump 11 from starting, even in the presence of the low RPV water level.

Conclusions:

Feedwater pump 11 will not start. The originally keyed answer of D is incorrect. Choice C is correct.

Question 12
Part 3 - Reference Documentation

A. Print of Powerboard 11 loads

B. Print of Powerboard 12 loads

C. Logic diagram for Feedwater Booster pump 12

D. Print of Powerboard 101 loads

E. Logic diagram for Feedwater pump 11

Question 15
Part 1 - Original Question

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000, A2.02	
	Importance Rating	2.8	

(K&A Statement) Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: High/low surge tank level

Proposed Question: Common 15

An unidentified leak from the RBCLC system has resulted in a loss of level in the Closed Loop Cooling (CLC) Makeup Tank in excess of makeup capability. The following conditions exist:

- Makeup Tank level is two (2) feet and slowly lowering.
- RBCLC Pump 12 is operating.
- RBCLC pressure is 38 psig and slowly lowering.
- RBCLC supply temperature is 93^{o}F and slowly rising.
- Operators have been dispatched to search for the location of the leak.

In accordance with N1-SOP-11, RBCLC Failure, which one of the following actions is required at this time?

A. Trip RBCLC Pump 12 and SCRAM the Reactor per N1-SOP-1.

B. SCRAM the Reactor per N1-SOP-1 and trip all Reactor Recirculation Pumps.

C. Trip RWCU pumps and initiate an Emergency Power Reduction per N1-SOP-1.1.

D. Initiate an Emergency Power Reduction per N1-SOP-1.1 and trip two Reactor Recirculation Pumps.

Proposed Answer: C.

Explanation (Optional):

C. Correct - N1-SOP-11 directs IF RBCLC cooling capability is challenged THEN trip RWCU pumps and the next step is Perform an Emergency Power Reduction per N1-SOP-1.1. With Makeup Tank level falling faster than makeup can provide and RBCLC supply temperature rising, cooling capability is challenged.

A. Incorrect - There is no direction to trip the remaining RBCLC Pump. This would compound the problem. Additionally SOP-11 requires a scram when all the RBCLC pumps have tripped and cannot be re-started. In this case RBCLC Pump 12 is operating.

B. Incorrect - This is directed in SOP-11 only if no RBCLC pumps are operating and none can be restarted.

D. Incorrect - There is no direction in SOP-11 to trip only two recirc pumps. All recirc pumps are tripped in the event that no RBCLC pumps are running and none can be started. This distractor action is similar to the direction to trip RWCU pumps to reduce heat load, but not correct.

Technical Reference(s): N1-SOP-11 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: N1101208000C01-RBO-10 (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 X
55.43 ____

Question 15
Part 2 - Justification

Analysis:

The stem conditions present two indications that RBCLC cooling capability is challenged. Both the low header pressure and high supply temperature conditions support the degraded cooling capability. This supports execution of the SOP-11.1 step to trip the RWCU pumps. The next step in the SOP-11.1 progression requires an Emergency Power reduction. These two actions support the originally keyed answer, C, as correct. SOP-11.1 is included in Part 3.

Additionally, the stem conditions present the candidates with a situation similar to what they have seen in training simulator scenarios. Based on their experiences with degraded RBCLC situations, the candidates believed that the lowering makeup tank level and lowering system pressure were signs that the RBCLC pumps were no longer effectively performing their design function. Having no RBCLC pumps fulfilling the design function is analogous to having no pumps running. In accordance with the station operating philosophy, as supported by both the Operations Manual and the training the candidates have received, it is desirable to operate in a conservative manner, taking action to operate the plant controls before conditions degrade to the point where automatic actions occur. Applicable sections of the Operations Manual are included in Part 3. Based on this operating philosophy of conservative decision making, and the belief that the RBCLC pumps were no longer fulfilling the design function, the override in SOP-11.1 to scram the reactor is deemed appropriate. This makes choice B also a correct answer.

Conclusions:

Both answers B and C are correct.

Recommendation:

Change the answer key to accept both B and C as correct answers.

Question 15

Part 3 - Reference Documentation

A. N1-SOP-11.1, RBCLC Failure

B. Operations Manual sections

Question 29
Part 1 - Original Question

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	286000 K3.03	
	Importance Rating	3.6	

(K&A Statement) Knowledge of the effect that a loss or malfunction of the FIRE PROTECTION SYSTEM will have on following: Plant protection

Proposed Question: Common 29

Which one of the following results from placing the Fire Zone Disconnect Switch in DISCONNECT for a zone where automatic fire suppression is available?

	Fire Alarms	Automatic Suppression
A.	Available	Available
B.	Available	Defeated
C.	Defeated	Available
D.	Defeated	Defeated

Proposed Answer: D.

Explanation (Optional):

D. Correct - In the Disconnect Mode, the fire alarm and actuation capabilities for a zone are both defeated. This may be desired to prevent inadvertent alarm or actuation during operations such as welding or cutting work.

A. Incorrect – Both alarm and actuation are defeated.

B. Incorrect – Both alarm and actuation are defeated.

C. Incorrect – Both alarm and actuation are defeated.

Technical Reference(s): N1-OP-21E, Sect. B and H.1.0 (Attach if not previously provided)

Attachment

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-286000-RBO-05 (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 X
55.43 ____

Question 29
Part 2 - Justification

Analysis:

The question asks for the effect of placing a single automatic fire suppression zone disconnect switch in DISCONNECT.

For the majority of the automatic fire suppression zones, the originally keyed answer, D, is correct, as justified in the original question. However, additional technical information has been discovered that supports two other answer choices as correct for certain automatic fire suppression zones.

The plant contains Wet Pipe sprinkler systems which use piping filled with pressurized water and are activated by fusible sprinklers. When a fire occurs, the heat produced will fuse a sprinkler causing water to flow. Placing a fire detection zone in DISCONNECT for this type of system will not prevent the sprinkler head from fusing in the event of a fire. Therefore, automatic suppression is not defeated by placing the zone disconnect switch in DISCONNECT.

The plant also contains Dry Pipe sprinkler systems which use a control valve to separate system water supply from the air-filled system piping. When the air pressure in the dry system has dropped (from the fusing of an automatic sprinkler) to the tripping point of the valve, the floating valve member assembly (air plate and water clapper) is raised by water pressure under the clapper. Water then flows into the intermediate chamber, overcoming the valve differential. Placing a fire detection zone in “DISCONNECT” will not prevent the sprinkler head from fusing in the event of a fire, and water flow will occur from the sprinkler system. Therefore, automatic suppression is not defeated by placing the zone disconnect switch in DISCONNECT.

Sections of N1-OP-21A, Fire Protections System – Water, are included in Part 3 to support the above discussion. In these instances, choice C is a correct answer.

Additionally, there are instances (i.e. Zone DA-2092 East and West) where there is more than one Zone Disconnect switch within the local fire panel for a given zone that must be placed in disconnect in order for the alarm and suppression function to be disabled. As seen on print F-39648-C SH 12, the split “subzone” configuration for zones DA-2092E and DA-2092W each are shown with 2 “Zone Disconnect” switches (S9 and S11 for the East zone and S13 and S15 for the West). The layout exists due to power supply limitations based on line/resistance loses tied to the number of detectors and length of wire etc. required to protect the area (physical size of the zone design considerations.) In the event that all of the switches are not placed in the disconnected position, alarm and suppression function would not be defeated. Part 3 includes print F-39648-C SH 12 and a screen view from the clearance tagging program which illustrates the need to operate more than one disconnect switch for certain systems.

Given the stem stating a singular zone disconnect switch being placed in DISCONNECT, these fire suppression zones with multiple disconnect switches would not be inhibited. In these instances, choice A is a correct answer.

Conclusions:

Given lack of stem focus as to the particular type of automatic fire suppression zone, choices A, C and D are all correct answers based on the various types of available suppression systems.

Recommendation:

Delete question 29 from the examination.

Question 29
Part 3 - Reference Documentation

A. Section of N1-OP-21A, Fire Protection System - Water

B. F-39648-C SH 12

C. Screen view from clearance tagging program

Question 67
Part 1 - Original Question

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	2.1.23	
	Importance Rating	4.3	

(K&A Statement) Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: Common 67

The plant was operating at 100% power when the following occurred:

- A loss of condenser vacuum resulted in a power reduction and reactor scram.
- Twelve (12) control rods failed to fully insert and are stuck at position 04.

Which one of the following methods is used to insert these rods, in accordance with station procedures?

A. Manually initiate Alternate Rod Insertion (ARI).

B. Reset the scram and insert a second manual scram.

C. Drive the control rods in to 00 using the Emergency In switch.

D. Place the mode switch in REFUEL and drive the rods in normally.

Proposed Answer: D.

Explanation (Optional):

D. Correct – Per EOP-SAP Bases all control rods are inserted to position 04 or less so there is no direction to enter EOP-3. With the control rods still not fully inserted, SOP-1 directs use of N1-OP-5 Control Rod Drive Sys, Sect. H.23. Control Rod Insertion (Shutdown Condition Hot). This section has the operator place the mode switch in refuel and insert the control rods using the normal Control Rod Movement switch.

A. Incorrect – There is no entry to EOP-3 therefore there is no direction to initiate ARI.

B. Incorrect – There is no entry to EOP-3 therefore there is no direction to reset the scram and insert another scram.

C. Incorrect – There is no procedure reference for this and no rods can be selected until the mode switch is placed in REFUEL. OP-5 specifically directs use of the normal Control Rod Movement switch, not the Emergency In switch.

Technical Reference(s): EOP-SAP Bases (Attach if not previously provided)
N1-SOP-01
N1-OP-05, Section H.23

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201001-RBO-10 (As available)

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

Question History: Last NRC Exam No

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

10 CFR Part 55 Content: 55.41 X
55.43

Question 67
Part 2 - Justification

Analysis:

The stem conditions present a reactor scram from 100% power with 12 control rods stuck at position 04. As included in the original justification, this leads to entry to N1-EOP-2, RPV Control, but does not warrant entry into N1-EOP-3, Failure to Scram. Entry to N1-EOP-3 is not warranted since all control rods are inserted to at least the Maximum Subcritical Banked Withdrawal Position of 04. The applicable portion of N1-EOP-2 and the EOP basis for Maximum Subcritical Banked Withdrawal Position are included in Part 3.

With no entry to N1-EOP-3, the operators are required to use N1-SOP-1, Reactor Scram, to insert control rods. N1-SOP-1, included in Part 3, directs the operator to use N1-OP-5, Control Rod Drive System, section H.23 to manually insert the control rods. N1-OP-5 section H.23 is included in Part 3. N1-OP-5 section H.23 step 7 directs use of the normal Control Rod Movement switch to insert the control rods. This supports the originally keyed answer, D, as correct.

The original question justification did not address N1-OP-5 Precaution and Limitation 9, which states, "Emergency rod in…function of the RMCS may be used to provide continuous rod movement in…at the direction of the Reactor Engineer…". The full Precaution and Limitation is included in Part 3. Given the highly abnormal stem conditions of multiple control rods failing to fully insert on a reactor scram, it is very likely that Reactor Engineering would authorize use of the Emergency Rod In switch for control rod insertion. This switch bypasses the RMCS timer, allowing the control rod insertion to proceed without waiting for the timer settle function, which is unnecessary when driving control rods to the full in position.

Conclusions:

Both answers C and D are correct.

Recommendation:

Change the answer key to accept both C and D as correct answers.

Question 67
Part 3 - Reference Documentation

A. Section of N1-EOP-2, RPV Control

B. NER-1M-095, NMP1 Emergency Operating Procedures and Severe Accident Procedures (EOP/SAP) Basis Document, Section 3.9, Maximum Subcritical Banked Withdrawal Position

C. N1-SOP-1, Reactor Scram

D. N1-OP-5, Control Rod Drive System, Section H.23

E. N1-OP-5, Control Rod Drive System, Precaution and Limitation 9

Question 86
Part 1 - Original Question

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	300000, A2.01	
	Importance Rating		2.8

(K&A Statement) Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Air dryer and filter malfunctions

Proposed Question: SRO 86

The plant has scrammed following a loss of all 115KV and 345KV offsite power. Partial electrical power is being provided from the Bennett's Bridge generator. The following conditions exist 5 minutes after the initial power loss:

- RPV water level is being maintained by HPCI.
- RPV pressure is being controlled using the Emergency Condensers (ECs).
- Instrument Air (IA) Receiver #12 pressure is 0 psig.
- IA Receiver #11 and Containment Spray System Air Test Receiver pressures are 100 psig.
- IA 94-19, BV – HSA RECEIVER TO IA SEPARATOR, is closed.
- An NAO dispatched to TB el. 291' reports that with both IA Filters in service the d/p on the filters is 20 psid and rising rapidly.

Which one of the following actions is required?

Prevent the loss of…

A. HPCI by directing the High Level Trip Bypass switch placed in BYPASS using Section H.3.0 of N1-OP-16, Feedwater Booster Pump to the Reactor.

B. IA by directing the operator to manually bypass IA Dryer 94-168 and 94-169 using Section H.3.0 of N1-OP-20, Service, Instrument and Breathing Air Systems.

C. HPCI by directing manual make up to the main condenser hotwell in accordance with N1-SOP-20.1, using Section H.13.0 of N1-OP-16, Feedwater Booster Pump to the Reactor.

D. IA by directing the operator to manually open IA-94-19, air systems crosstie valve in accordance with N1-SOP-20.1, using Section H.2.0 of N1-OP-20, Service, Instrument and Breathing Air Systems.

Proposed Answer: C.

Explanation (Optional):

C. Correct - If a LOOP occurs the IA Compressors are lost and system components fail to a safe position. Sufficient air is available for approximately 15 minutes in the IA piping volume and Containment Spray Air Test Receiver in order to implement EOP actions and maintain safe plant conditions; In this case the Bennett's Bridge generator can supply some selective loads including HPCI. However IA is being lost because of a high d/p on the IA filters from IA Receiver # 11 and the Containment Spray System Air Test Receiver (the largest reservoir of IA). This loss of IA will result in the hotwell level control valves failing closed. SOP-20.1 directs using OP-16 to manually make up to hotwell to supply the HPCI system.

A. Incorrect – There is no direction or need to place the High Level Trip Bypass Switch in BYPASS. If HPCI flow becomes a problem because of the air failure because the FCVs fail as is (lock up) on loss of IA they do not need to be immediately operated manually. RPV water level can be controlled for the short term by alternately starting and stopping the HPCI Pump until the FCVs are placed in manual.

B. Incorrect – OP-20 does contain a section for bypassing the dryers if necessary to maintain IA pressure, however bypassing the IA Dryers will NOT bypass the filters.
There is no manual (or auto) bypass around the Instrument Air Filters

D. Incorrect - Opening IA-94-19, air systems crosstie valve will not restore IA pressure because the crosstie is upstream of the IA filters. SOP-20.1 directs verifying that IA-94-19 opens but does NOT direct manually opening the valve.

Technical Reference(s): N1-OP-20, Sect B.
N1-SOP-20.1
C18011-C
N1-OP-16, Section B and H.13.0
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-278001-RBO-10 (As available)

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

Question History: Last NRC Exam No

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

Question 86
Part 2 - Justification

Analysis:

The stem conditions present a degrading Instrument Air (IA) system due to a rising differential pressure across the IA Filters. The question asks which one of the listed actions will prevent loss of the associated system. The originally keyed answer, C, presents that the loss of HPCI is prevented solely by taking manual control of Hotwell level. In accordance with N1-SOP-20.1, Instrument Air Failure, both manual control of Hotwell level and manual control of the Feedwater flow control valves is required to preserve HPCI. N1-SOP-20.1 is included in Part 3. Manual control of Hotwell level alone will preserve the water source, but not allow injection of this water to the RPV. With the stem focus on preservation of the entire HPCI function, choice C does not present a correct answer.

Given choice C as invalid, the best choice is B. Given a degrading instrument air system with indications of rising system flow restriction, bypassing of the IA Dryers is a reasonable choice to lower total system flow restriction and preserve the ability to supply air pressure to loads.

Conclusions:

The originally accepted answer, C, is not technically valid given the stem focus. Choice B is the remaining reasonable answer choice.

Recommendation:

Change the answer key to accept choice B as correct and make choice C incorrect.

Question 86

Part 3 - Reference Documentation

A. N1-SOP-20.1, Instrument Air Failure

ATTACHMENT 3

NRC RESOLUTION OF FACILITY POST EXAM COMMENTS

SUMMARY

Question Number	Licensee Post-Exam Comments	NRC Responses
Q 2	Delete	Question Deleted
Q 12	Change correct answer to “C”	Answer Changed
Q 15	Add second correct answer “B”	Denied
Q 29	Delete	Question Deleted
Q 67	Add second correct answer “C”	Accept “C” and “D”
Q 86 (SRO 11)	Change correct answer to “C”	Question Deleted

Question 2

NRC Resolution:

Comment accepted, question deleted. The key answer was derived from the EOP basis for performing reactor pressure vessel (RPV) blowdown before exceeding the heat capacity temperature limit (HCTL) curve. The examiner reviewed the basis document and determined that basis includes the assumption that torus cooling is unavailable. This condition was not provided in the question stem, and as a result there is no correct answer.

Question Common 12

Facility Comment:

Change correct answer from “D” to “C”. The key answer is technically incorrect

NRC Resolution:

Comment accepted, answer changed. This question asks how the plant responds to an electrical transient. The examiners reviewed system lesson information submitted by the facility and noted that the undervoltage trip of condensate booster pumps and interlocks between condensate booster pumps and feedwater pump 11 support the requested answer change.

Question Common 15

NRC Resolution:

Comment not accepted. The question asks for the procedurally required action for the stem conditions of a lowering RBCLC failure, and the answer to that question is “C” only.

Question Common 29

NRC Resolution:

Comment accepted, question deleted. The facility provided fire protection system description information concerning wet pipe and dry pipe sprinkler systems, DC power actuated CO2 systems, and systems with more than one disconnect switch. Since the question does not specify any particular type of system, distractors “ A”, “C” and “D” are potentially correct and mutually exclusive answers.

Question 67

NRC Resolution:

Comment accepted. The facility referenced N1-OP-5 precaution and limitation 9, which supports the requested second answer.

Question 86 (SRO 11)

Change correct answer

NRC Resolution:

Comment partially accepted, question deleted.

The facility is correct in that the procedurally required action for the stem conditions is to take local control of both hotwell makeup AND feedwater flow control valves. However, the suggested alternate answer is NOT a procedurally directed action for the high instrument air filter d/p condition in the question stem and would not be effective. Accordingly, this question has no correct answer and is deleted.