UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE RD. SUITE 210
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February 7, 2017

Mr. Peter A. Gardner
Senior Vice President and CNO
Monticello Nuclear Generating Plant
Northern States Power Company, Minnesota
2807 West County Road 75
Monticello, MN 55362–9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT – NRC INITIAL LICENSE EXAMINATION REPORT 05000263/2016301

Dear Mr. Gardner:

On December 29, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed the initial operator licensing examination process for license applicants employed at your Monticello Nuclear Generating Plant. The enclosed report documents the results of those examinations. Preliminary observations noted during the examination process were discussed on November 17, 2016, with you and other members of your staff. An exit meeting was conducted by telephone on January 5, 2017, between Mr. G. Allex of your staff and Mr. D. Reeser, Chief Operator Licensing Examiner, to review the proposed final grading of the written examination for the license applicants. During the telephone conversation, NRC resolutions of the station's post examination comments, initially received by the NRC on December 2, 2016, were discussed.

The NRC examiners administered an initial license examination operating test during the week of November 14, 2016. The written examination was administered by NRC examiners and Monticello Nuclear Generating Plant department personnel on November 18, 2016. Four Senior Reactor Operator and one Reactor Operator applicants were administered license examinations. The results of the examinations were finalized on December 29, 2016. One applicant failed one or more sections of the administered examination and was issued a proposed license denial letter. Four applicants passed all sections of their respective examinations and two were issued senior operator licenses and one was issued an operator license. In accordance with NRC policy, the license for one senior operator license applicant is being withheld pending the outcome of any written examination appeal that may be initiated.

The administered written examination and operating test, as well as documents related to the development and review (outlines, review comments and resolution, etc.) of the examination will be withheld from public disclosure until January 1, 2019. However, if an applicant received a proposed license denial letter due to unsatisfactory performance on one or more portions of the examination, that applicant will receive a copy of the applicable portions of the examination. For examination security purposes, your staff should consider those examination materials uncontrolled and exposed to the public.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Robert J. Orlikowski, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50-263
License No. DPR-22

Enclosures:
1. OL Examination Report 05000263/2016301
2. Post Exam Comments, Evaluation, and Resolutions
3. Simulation Facility Fidelity Report

cc: Distribution via LISTSERV®
P. Kissinger, Training Manager,
Monticello Nuclear Generating Plant

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-263
License No:	DPR-22
Report No:	05000263/2016301
Licensee:	Northern States Power Company, Minnesota
Facility:	Monticello Nuclear Generating Plant
Location:	Monticello, MN
Dates:	November 14, through December 29, 2016
Inspectors:	D. Reeser, Operations Engineer; Chief Examiner R. Baker, Operations Engineer; Examiner B. Palagi, Senior Operations Engineer; Examiner
Approved by:	R. Orlikowski, Chief Operations Branch Division of Reactor Safety

SUMMARY

Examination Report 05000263/2016301; 11/14/2016 – 12/29/2016; Northern States Power Company, Minnesota, Monticello Nuclear Generating Plant; Initial License Examination Report.

The announced initial operator licensing examination was conducted by regional Nuclear Regulatory Commission examiners in accordance with the guidance of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 10.

Examination Summary

Four of five applicants passed all sections of their respective examinations. Two applicants were issued senior operator licenses and one applicant was issued an operator license. One applicant failed one or more sections of the administered examination and was issued a proposed license denial. The license for the remaining applicant is being held and may be issued pending the outcome of any written examination appeal. (Section 4OA5.1).

REPORT DETAILS

4OA5 Other Activities

.1 Initial Licensing Examinations

a. Examination Scope

The U.S. Nuclear Regulatory Commission (NRC) examiners and members of the facility licensee's staff used the guidance prescribed in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 10, to develop, validate, administer, and grade the written examination and operating test. NRC examiners prepared the outline and developed the written examination and operating test. The NRC examiners validated the proposed examination during the week of October 17, 2016, with the assistance of members of the facility licensee's staff. During the on-site validation week, the examiners audited one license application for accuracy. The NRC examiners, with the assistance of members of the facility licensee's staff, administered the operating test, consisting of job performance measures (JPMs) and dynamic simulator scenarios, during the period of November 14 through 17, 2016. The NRC examiners, with the assistance of members of the facility licensee's staff, administered the written examination on November 18, 2016.

b. Findings

(1) Written Examination

During the validation of the written examination, several questions were modified or replaced. Changes made to the written examination were documented on Form ES-401-9, "Written Examination Review Worksheet," which will be available in 24 months electronically in the NRC Public Document Room or from the Publicly Available Records component of Agencywide Document Access and Management System (ADAMS).

On December 2, 2016, the licensee submitted documentation noting that there were 12 post-examination comments for consideration by the NRC examiners when grading the written examination. The post-examination comments and the NRC resolution for the post-examination comments are included as Enclosure 2 to the report. The proposed NRC-developed written examination, the written examination outlines and worksheets, as well as the final as-administered examination and answer key (ADAMS Accession Number ML 17023A123), will be available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's ADAMS in January 2019.

The NRC examiners graded the written examination on December 29, 2016, and conducted a review of each missed question to determine the accuracy and validity of the examination questions.

(2) Operating Test

During the validation of the operating test, minor modifications were made to several JPMs, and some minor modifications were made to the dynamic simulator scenarios. Changes made to the operating test, documented in a document titled, "Operating Test

Comments," the proposed NRC-developed dynamic simulator scenarios, JPMs, and associated operating test outlines, as well as the final as-administered dynamic simulator scenarios and JPMs, will be available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's ADAMS in January 2019.

The NRC examiners completed operating test grading on December 29, 2016.

(3) Examination Results

Four applicants at the Senior Reactor Operator level and one applicant at the Reactor Operator level were administered written examinations and operating tests. Three applicants passed all portions of their examinations and were issued their respective operating licenses on December 29, 2016.

One applicant failed one or more sections of the administered examination and was issued a proposed license denial. One applicant passed all portions of the license examination, but received a written test grade below 82 percent. In accordance with NRC policy, the applicant's license will be withheld until any written examination appeal possibilities by other applicants have been resolved. If the applicant's grade is still equal to or greater than 80 percent after any appeal resolution, the applicant will be issued an operating license. If the applicant's grade has declined below 80 percent, the applicant will be issued a proposed license denial letter and offered the opportunity to appeal any questions the applicant feels were graded incorrectly.

.2 Examination Security

a. Scope

The NRC examiners reviewed and observed the licensee's implementation of examination security requirements during the examination validation and administration to assure compliance with Title10 of the *Code of Federal Regulations*, Section 55.49, "Integrity of Examinations and Tests." The examiners used the guidelines provided in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," to determine acceptability of the licensee's examination security activities.

b. Findings

None

4OA6 Management Meetings

.1 Debrief

The chief examiner presented the examination team's preliminary observations and findings on November 17, 2016, to Mr. P. Gardner, Senior Vice President, and other members of the Monticello Nuclear Generating Plant Operations and Training Department staff.

.2 Exit Meeting

The chief examiner conducted an exit meeting on January 5, 2017, with Mr. G. Allex, General Supervisor Operation Training by telephone. The NRC's final disposition

of the station's post-examination comments were disclosed and discussed with Mr. Allex during the telephone discussion. The examiners asked the licensee whether any of the material used to develop or administer the examination should be considered proprietary. No proprietary or sensitive information was identified during the examination or debrief/exit meetings.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Becker, Operations Training Instructor – Examination Lead
M. Peterson, Xcel Fleet Training
B. Koenig, Operations Shift Manager/ILT Supervisor
G. Allex, General Supervisor Operations Training
C. Peterson, Operations Training Supervisor
P. Kissinger, Training Manager

U.S. Nuclear Regulatory Commission

P. Zurawski, Senior Resident Inspector
D. Krause, Resident Inspector
D. Reeser, Chief Examiner
R. Baker, Examiner
B. Palagi, Examiner

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened, Close, and Discussed

None

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access and Management System
CFR	*Code of Federal Regulations*
JPM	Job Performance Measures
NRC	U.S. Nuclear Regulatory Commission

QUESTION No. 8

The plant is at rated conditions when an event occurs resulting in the common Service/Instrument air header pressure slowly lowering.

Assuming the common air header pressure continues to lower; which of the following alarms would be the FIRST to be received that ALSO requires entry into an AOP (Abnormal Operating Procedure)?

A. ROD DRIFT (5-A-27)
B. INST AIR HEADER LOW PRESS (6-B-34)
C. SERVICE AIR HEADER LOW PRESS (6-B-35)
D. SCRAM PILOT HEADER HI/LO PRESS (5-B-22)

Answer: **B**

DISTRACTOR ANALYSIS

A. Incorrect: at <60 psig, the control rods will start to drift and the reactor must be manually scrammed; but AOP shall be entered prior to.

B. **Correct**: when Air Header Pressure reaches 85 psig, annunciator 6-B-34 (INST AIR HEADER LOW PRESS) will alarm. ARP 6-B-34 directs AOP C.4-B.08.04.01 entry. Per AOP C.4-B.08.04.01, Table 1, if the IA pressure is reduced to 85 psig this is the first time operator AOP action will be required.

C. Incorrect: at 82 psig, Annunciator 6-B-35 (SERVICE AIR HEADER LOW PRESS) alarms and CV-1474, Serv Air Isol CV valve, closes; AOP entry is not directly required.

D. Incorrect: Annunciator 5-B-22 (SCRAM PILOT HEADER HI/LO PRESS) alarms at 60 psig. AOP entry would not be required until control rods start to drift.

APPLICANT COMMENT/CONTENTION

The applicant contends that there are two correct answers, choices ‘A’ and ‘B’.

While alarm C.6-006-B-34 does direct entry into the AOP for Loss of Instrument Air; the question specifies an entry into an AOP; Rod Drift which is not expected until < 60# could not only occur prior to the Loss of Instrument Air, but is also a different AOP Entry from the initial AOP as the initial Conditions State that we are in a Loss of Instrument Air AOP; inferred as already in a loss of Instrument Air and therefore an additional AOP entry could be actuated. Rod Drifting (C.6-005-A -27) for any other reason is a correct answer as it directs entry into Control Rod Drifting AOP C.4-B.01.03.C. Thus, there are (2) correct answers:

a) Rod Drift C.6-005-A-27 directing entry into C.4-B.01.03.C, and
b) Inst Air Header Low Press C.6-006-B-34 directs entry into C.4-B.08.04.01'A

QUESTION No. 8 (page 2 of 3)

FACILITY RESPONSE AND PROPOSED RESOLUTION

The facility response states that the question is acceptable as written.

> *Candidate should assume instrument air header pressure is at the normal value from the initial conditions in the stem. If air header pressure started at normal system pressure and continues to lower, 6-B-34 (INST AIR HEADER LOW PRESS) would be received at 85 psig and 5-A-27 (ROD DRIFT) would be received at approximately 60 psig. Both would require entry into an AOP but 6-B-34 would occur FIRST.*
>
> *Entry into the loss of instrument air AOP would be allowed at the discretion of the CRS with a lowering instrument air header pressure prior to receipt of the alarm. As stated above the first annunciator expected to be received on lowering pressure is 6-B-34 (INST AIR HEADER LOW PRESS). If the intellectual jump is made that the question requires discarding the first annunciator received as a correct answer because the AOP was already discretionarily entered (though not stated in the stem) then answer choice A would be the next annunciator that is expected to be received that requires entry into a different AOP.*
>
> *With the assumptions made by the candidate, answer choice A is a partially correct answer, however, the question asks "…entry into an AOP..." not entry into the next AOP or an AOP that has not already been entered. Thus, the best answer is still answer choice B.*
>
> *Clarification not requested during exam administration.*
>
> *Question acceptable as written.*
>
> *Reference:*
> *6-B-34 (INST AIR HEADER LOW PRESS)*
> *5-A-27 (ROD DRIFT)*
> *C.4-B.08.04.01.A (LOSS OF INSTRUMENT AIR)*

NRC EVALUATION/RESOLUTION

The applicant's interpretation of the initial conditions, given in the first sentence of the question stem, is not supported by the information provided. The only factual information that can be inferred from the provided information is that: (a) the common Service/Instrument air header pressure prior to the event was at rated conditions (i.e. normal); and (b) the common Service/Instrument air header pressure, after the event, is at a lower value and continuing to lower slowly. As stated in the facility's response, early entry into the "Loss of Instrument Air" abnormal response procedure (AOP) is permissible, that decision would be based on the current air header pressure and trend information; neither were provided, nor can be inferred by the provided information; therefore, to assume that the AOP has already been entered is unsupportable. Additionally, the question does not ask which AOP will be entered, but which of the listed alarms would be the FIRST to be received, that would ALSO require entry into an AOP. As stated above, the first alarm that is expected to be received is 6-B-34.

QUESTION No. 8 (page 3 of 3)

The associated alarm response procedure (ARP) directs the operator to enter C.4-B.08.04.01.A (Loss of Instrument Air). The associated alarm is also listed in the AOP as an indication (i.e. "entry condition") of a loss of Instrument Air. Alarm 6-B-35 is also listed in the AOP as an indication of a Loss of Instrument Air, but is not expected to be received until after 6-B-34. The ARP for 6-B-35 does not specifically direct entry into the Loss of Instrument Air AOP.

CONCLUSION

Based the information provided and a review of the applicable references, the NRC concludes that the question is acceptable as written, and that the original answer is the only correct answer.

QUESTION No. 13

The plant was at rated conditions when a SBO and ELAP event occurred. Given the following:

- The Heat Capacity Limit has been exceeded
- RPV water level is being maintained with HPCI & RCIC
- The CRS has entered C.5-2002 (EMERGENCY DEPRESSURIZATION)

During the depressurization, the CRS directs a pressure band of 150-200 psig.

Is the directed pressure band correct? Why or why not?

A. CORRECT; ≥150 psig is required to maintain core cooling.
B. CORRECT; ≥150 psig is required to mitigate primary containment challenges.
C. INCORRECT; full depressurization is required to maintain core cooling.
D. INCORRECT; full depressurization is required to mitigate primary containment challenges.

Answer: **A**

DISTRACTOR ANALYSIS

A. **Correct**: Under circumstances requiring ED, it is generally desirable to full depressurize the RPV. However, while RPV pressure reductions will tend to increase flow from motor-driven pumps, full depressurization will result in loss of steam driven injection sources. RPV pressure reduction must be coordinated with core cooling strategies. Full depressurization is only appropriate if adequate core cooling will not be sacrificed as a result. In the ELAP condition, HPCI and RCIC will be the only injection sources so RPV pressure is to be maintained ≥ 150 psig.

B. Incorrect: The pressure band is not for containment concerns.

C. Incorrect: Full depressurization not required.

D. Incorrect: The pressure band is not for containment concerns and full depressurization not required.

APPLICANT COMMENT/CONTENTION

The applicant contends that there is no correct answer.

The choices offered don't offer the prescribed pressure band of 150 - 300 # IAW C.5-4000. Further, the assumption must be made that there is an Emergency Depressurization in progress which does offer the override to maintain pressure > 150 #, but one can assume that we are Depressurizing IAW the Center Leg of C.5-1100. Part H of C.5-1100 directs a blowdown when Low Capacity Injection Systems are lined up and High Capacity Injection Sources are Unavailable; thus, we would not blowdown if RCIC was in service and Injecting. Instead, we would maintain -40" to +100" and 150 - 300# using RCIC and Alternate Depressurization Methods per C.5-1100. In either case; the band IAW C.5-4000 is 150 - 300#; thus, no correct answer exists.

QUESTION No. 13 (page 2 of 4)

****In further review of the question, the blowdown is prescribed for exceeding the Heat Capacity Limit; while important; this is contrary to the guidance in the C.5-4000 which prescribes a Blowdown when:*

a) RCIC and HPCI are no longer available, or
b) Diesel driven pump lined up and ready for injection, or
c) ELAP declared, do not wait for RPV level to reach -126".

Additionally, Primary Containment venting addresses Torus temperature>2I2F and > 10# Drywell pressure respectively.

Given the initial conditions and the directed band of 150 - 200#, there is no correct answer per C.5-4000, C.5-1100, or C.5-1200.

One would require more parameters such as RPV water level, status of other injection sources, as well as if there was other EOP entries that may require a blowdown to establish a basis as the Blowdown for Heat Capacity Limit is defined per C.5.1-1200 ensures the highest Torus Temperature that a Blowdown will not exceed Torus Design Temperature, or Drywell Pressure Limit. Therefore, the Blowdown precludes Loss of Primary containment Integrity and Pressure suppression Function.

FACILITY RESPONSE AND PROPOSED RESOLUTION

The facility response states that the question is acceptable as written.

When Emergency Depressurization is required during an ELAP with only HPCI and RCIC maintaining RPV level, the vessel should not be fully depressurized. RPV pressure should be controlled as low as possible >150 psig using Alternate Depressurization Systems (RCIC & HPCI) while maintaining adequate core cooling.

ELAP HPCI Only Pressure Band: 150-1000 psig (C.5-4000)
ELAP RCIC & HPCI Pressure Band: 150-300 psig (C.5-4000)

The pressure band listed in the stem of the question does not exactly match the bands stated above in C.5-4000, however, it is fully within the above bands and though difficult to maintain it would be acceptable pressure band for the indicated conditions.

The question specifically asks if the pressure band is "...correct..." and thus an interpretation had to be made as to whether an "acceptable" pressure band is also a "correct" pressure band. This could lead to some confusion on the part of the candidate.

However, the second part of both answer choices C & D is absolutely incorrect as it states that full depressurization is required. This contradicts the reason for believing that the initial pressure band was incorrect; the upper limit was not high enough (300 vs. the listed 200). Answer choices A & B both state that pressure be maintained >150 psig which is the most limiting factor in this plant condition.

QUESTION No. 13 (page 3 of 4)

Question clarification was not requested during administration.

Question acceptable as written.

Reference:
C.5-4000 (SBO)
C.4-B.09.02-A (SBO)
C.5-1200 (Primary Containment Control)
C.5-2002 (Emergency RPV Depressurization)

NRC EVALUATION/RESOLUTION

The applicant's contention initially tried to make the argument that the "depressurization," mentioned in the second paragraph of the question stem (i.e. the sentence following the initial conditions) could be assumed, to be due to actions being taken in accordance with C.5-1100, and that actions required by C.5-2002 had not yet been initiated. The question stem clearly states that the CRS had entered C.5-2002. The very first item in C.5-2002 is a NOTE which includes the statement: "This procedure overrides the RPV pressure control actions in: C.5-1100, "RPV Control" (Pressure section); [and] C.5-2007, "Failure to Scram" (Pressure section)." Therefore, the applicant's assumption is not supported by the information contained in the question stem, nor by the direction provided in C.5-2002.

In the second paragraph of the applicant's contention, the applicant implies: that the guidance in C.5-4000 supersedes the direction, in C.5-1200, to enter C.5-2002 due to the inability to maintain Torus temperature below the Heat Capacity Temperature Limit; and that the only time that C.5-2002 should be entered is when: (a) RCIC and HPCI are no longer available; or (b) a diesel driven pump lined up and ready for injection. The guidelines contained in C.5-4000, are provided to supplement the EOPs, not replace them. The EOP Strategies in C.5-4000 are provided to assist the operator in maintaining the availability of the steam driven injection sources, until AC power is available, other injection sources (e.g., diesel driven pumps) of sufficient capacity are available. If the EOPs direct entry into, and implementation of, C.5-2002, the procedure is required to be implemented, with the understanding that the full depressurization will be terminated if necessary to preserve core cooling. C.5.1-2002 states: "Full depressurization and cooldown is appropriate only if adequate core cooling will not be sacrificed as a result. Loss of adequate core cooling would compound the plant challenges requiring emergency depressurization and increase any resulting radioactivity release. Core cooling is thus prioritized over other EOP objectives. If, at any time during RPV depressurization, it is anticipated that continued pressure reduction will result in loss of injection flow required for adequate core cooling, the depressurization is terminated. Pressure is then controlled as low as practicable but above the minimum value at which the required injection flow can be sustained."

QUESTION No. 13 (page 4 of 4)

The RPV pressure and level control bands, specified in C.5-4000 and C.4-B.09.02A, are guidelines similar to the control bands specified in OWI-03.06 (Strategies for Successful Transient Mitigation). The OWI defines Operations personnel mitigation strategy expectations to ensure consistent implementation of Operator fundamentals for an effective response. The guidance contained in that instruction compliments operating procedures. It is NOT intended to replace or supersede approved operating procedures. The OWI identifies several control bands which vary depending on particular circumstances. The OWI also recognizes that the control bands may need to be adjusted dependent upon the specific conditions created by the transient. The pressure control band specified for the ELAP mitigation strategy is based on: (1) performing a controlled cooldown/depressurization to facilitate RPV water level control actions and reduce the containment heat-up if a blowdown is later required; (2) maintaining availability of steam driven injection systems (150 psig minimum pressure) as long as possible; and (3) prioritizing the desire to ensure that adequate core cooling is maintained, even if other EOP objectives have to be sacrificed. As mentioned earlier, the loss of adequate core cooling would compound the plant challenges requiring EOP entry and increase any resulting radioactivity release. While the pressure band given in the stem of the question does not exactly match the recommended control band given in C.5-4000 or C.4-B.09.02.A and may not be optimum, it falls within the bounding requirements identified above and is therefore acceptable.

CONCLUSION

Based the information provided and a review of the applicable references, the NRC concludes that the question is acceptable as written, and that the original answer is the correct answer.

QUESTION No. 30

The plant is in MODE 4 with the 12 RHR Pump operating in a Normal Shutdown Cooling Mode. The desired RCS temperature control band is 120-140°F.

- Both Reactor Recirculation pumps are OFF
- The CRD and RWCU systems are SHUTDOWN

Time	Recirc Loop A Suction Temperature	Recirc Loop B Suction Temperature	12 RHR HX Inlet Temperature	RWCU Inlet Temperature
0900	141	139	140	139
0915	140	139	138	140
0930	140	138	137	139
0945	139	139	135	139
1000	138	138	134	140

Give the above information, what is the RCS Heatup/Cooldown rate.

A. Cooling down at 3°F/hr
B. Cooling down at 1°F/hr
C. Cooling down at 6°F/hr
D. Heating up at 1°F/hr

Answer: **C**

DISTRACTOR ANALYSIS

A, B, and D are incorrect but plausible variations of actions in order to decrease the recirculation loop temperature at a slower rate (i.e., decrease cooldown rate).

A. Incorrect: There may be some back-flow through the 11 RR Loop, but the temperature of the loop will not be representative of the RCS temperature.

B. Incorrect: With the 12 RR Loop suction or discharge valve shut there will be no flow through the loop and the indicated temperature will not be representative of the RCS temperature.

C. **Correct**: The coolant flowing throught the in-service RHR train will be representative of the RCS temperature.

D. Incorrect: With the RWCU system shutdown there will be no flow through the system and the indicated temperatue will not be representative of the RCS temperature.

QUESTION No. 30 (page 2 of 3)

APPLICANT COMMENT/CONTENTION

The applicant contends that answer choice 'A' is the correct answer.

I believe Question 30 should accept Answer A (cooling down at 3°F/hr) as the correct answer. Per [procedure] 0118 (Reactor Vessel Temperature Monitoring) Recirc Loop A is included as an acceptable method when verifying cooldown rates.

- *Since idle loop injection is occurring, the 12 Recirc [Pump] Disch valve will be closed (see B.03.04-05, page 36)*
- *Since both Recirc pumps are secured, normal system shutdown of both Recirc pumps will have implemented. This procedure indicates that the suction and discharge valves for 11 Recirc will be open to have the same cooldown rate as the vessel (see B.01.04-05, page 57).*
- *Also the 0118 indicates that RPV-508 (RPV rate of change), which only includes input from RHR, is to be used to aid in assuring cooldown rates; implying that actual cooldown should be used from loop A temperature.*

FACILITY RESPONSE AND PROPOSED RESOLUTION

The facility response states that the question is acceptable as written.

IAW 2204 (PLANT SHUTDOWN); RPV coolant change in temperature on Screen 505 and Computer points RPV508 (RPV Rate of Change) and RPV802 (RPV Temperature) from Special Log 15 are the representative data points for monitoring coolant temperature rate of change as required by Tech Specs.

Based on the way that Screen 505 and Computer point RPV802 are validated, once shutdown cooling is put in service 100% of the temperature input is from RHR. With Recirc Pumps secured, Recirc loop temperatures no longer provide input to RPV802.

Question acceptable as written.

Reference: 2204 (Plant Shutdown)

NRC EVALUATION/RESOLUTION

The focus of the question is on whether or not the applicant understands the physical processes of how temperature instrumentation, remotely located in fluid systems connected to the Reactor Pressure Vessel, can be used to indirectly measure Reactor Coolant System (RCS) temperature. The question does not provide for specific computer point, recorder, or other indicator names or identification numbers, nor does it ask for that information. The key element to answering the question is the knowledge that for a temperature instrument to provide an output that is representative of the RCS temperature, there has to be a flow path from the RCS, past the temperature element in the connected system. The higher the circulation flow rate through the RPV and the connected system, the more representative the indication will be of the average RCS temperature.

QUESTION No. 30 (page 3 of 3)

The applicant contends that since the 11 Recirculation Loop is not expected to be isolated, the associated temperature monitor would be representative of the RCS heatup/cooldown rate. The applicant bases his contention on a NOTE in the Reactor Recirculation System operating procedure that states: "The pump suction and discharge valves are to be left open to facilitate recirculation loop cooldown at the same rate as the reactor vessel cooldown." However the step immediately following the NOTE only partially opens (three second stroke; a small fraction of the total stroke time) the discharge valves. This would permit only a small fraction of the flow through the RPV to be diverted through the loop. Even if the discharge valve were to be fully opened, the flow in the 11 Recirculation Loop would be in the reverse direction through the discharge piping and would be restricted by the Jet Pump nozzle openings and the idle pump. The majority of the flow from the RPV would be drawn through the 11 Recirculation Loop suction line to the 12 RHR Pump suction. Additionally, as stated in the question stem, the plant is already in Mode 4, the plant cooldown is complete, and the concern addressed by the referenced NOTE is of minor concern since the temperature difference between the RCS and the un-isolated Recirculation system loop will be minor.

Section IV of procedure C.3 (Shutdown Procedure) specifies the minimum requirements necessary to ensure that proper Reactor conditions are maintained in MODES 4 and 5, and includes guidance for monitoring key reactor parameters. The guidance for monitoring Reactor Water Temperature specifies that:

If neither Recirculation Pump is running.

Then coolant temperature should be obtained from one of the following:

1) *If RHR Shutdown Cooling is in service,*
   Then coolant temperature should be obtained from the RHR Heat Exchanger inlet temperature.
2) *If the RWCU system is in service,*
   Then coolant temperature should be obtained from the RWCU inlet temperature.

Knowledge of the physical relationships and interactions discussed above, as well as the procedure guidance contained in C.3 (Shutdown Procedure), clearly support that answer choice 'C' is the correct answer. The contention by the applicant that answer choice 'A' is the only correct answer, or that it is also correct is not support by information provided above. While the 11 Recirculation Loop temperature may trend similarly, the temperature information will not be truly representative of conditions with the RPV.

CONCLUSION

Based the information provided and a review of the applicable references, the NRC concludes that the question is acceptable as written, and that the original answer is the correct answer.

QUESTION No. 33

An ATWS event has occurred and the OATC has initiated SBLC from C-05.

Complete the statements below.

Forced circulation is __(1)__ to ensure adequate dispersion of the boron solution into the core.

RWCU pumps __(2)__ receive an automatic trip signal.

A. (1) required
   (2) will

B. (1) required
   (2) will NOT

C. (1) NOT required
   (2) will

D. (1) NOT required
   (2) will NOT

Answer: **C**

DISTRACTOR ANALYSIS

A. (1) Incorrect, without a recirculation pump running, natural circulation provides adequate dispersion of the solution into the core.
   (2) Correct

B. (1) Incorrect, without a recirculation pump running, natural circulation provides adequate dispersion of the solution into the core.
   (2) Incorrect, RWCU pumps trip on SBLC system actuation unless the isolation interlocks are bypassed per the EOP.

C. **Correct**

D. (1) Correct
   (2) Incorrect, RWCU pumps trip on SBLC system actuation unless the isolation interlocks are bypassed per the EOP.

QUESTION No. 33 (page 2 of 3)

APPLICANT COMMENT/CONTENTION

The applicant implies in their comment that answer choice 'A' is also a correct answer.

> *Candidate understood "forced" circulation to include [either] natural or motor driven circulation. In either case, a force must be imposed on the water, whether this force is from a difference in density or driven by a pump, there always exists a force on the water, or it would not move. Therefore, in order to distribute the Boron the operator "forces" circulation by changing level.*

FACILITY RESPONSE AND PROPOSED RESOLUTION

The facility response states that the question is acceptable as written.

> *Procedure B.03.05-05 (SBLC): The difference between forced recirculation and natural circulation is defined as follows: "Without a Recirculation pump running, natural circulation provides adequate dispersion of the solution into the core."*
>
> *Procedure C.5.1-2007 (Failure to Scram): With natural circulation flow reduced, the boron injected by SBLC may simply collect in the lower plenum and not reach the core until flow is reestablished. However, once enough boron is injected, RPV water level is raised to reestablish natural circulation flow and distribute the boron throughout the core region.*
>
> *MNGP originally revised this question to state "Recirc flow is…" and the NRC requested it be changed back to "Forced circulation is..."*
>
> *Clarification on the meaning of "Forced Circulation" was not requested during the administration of the exam.*
>
> *Question Acceptable as Written.*
>
> *Reference:*
> *B.03.05-05 (SBLC)*
> *C.5.1-2007 (Failure to Scram)*

NRC EVALUATION/RESOLUTION

Knowledge and usage of the Thermal Hydraulic terms "natural circulation" and "forced circulation" are fundamental principles that applicants are expected to be very familiar with. The term "forced circulation" is generally understood to be circulation that is driven by mechanical forces, as opposed natural forces such as gravity and buoyancy. The applicant's statement that they understood "forced" circulation to be flow driven by "any" force is inconsistent with generic operating fundamental knowledge.

QUESTION No. 33 (page 3 of 3)

C.5.1-2007 (Failure to SCRAM) describes the strategies for mitigating an ATWS event. This initial response includes, termination of “forced circulation” (runback and trip of the Reactor Recirculation pumps) and lowering of RPV level, to reduce sub-cooling and increase voiding within the core to rapidly reduce power, until control rod insertion and/or Boron injection can be implemented to shutdown the reactor. Forced circulation is not restored until after the C.5-2007 is exited. If Boron is injected, RPV water level is maintained low until enough Boron has been injected to maintain hot shutdown conditions within the reactor. RPV level is then slowly raised to establish “natural circulation” for mixing and circulating the borated reactor coolant within the core while additional Boron is injected until cold shutdown conditions are achieved.

The applicant’s contention cannot be supported, given the fundamental nature of the terms discussed above and the information provided in the EOP basis.

CONCLUSION

Based the information provided and a review of the applicable references, the NRC concludes that the question is acceptable as written, and that the original answer is the correct answer.

QUESTION No. 39

The plant was at rated conditions with HPCI out of service when a Group I Isolation occurred. Given the following:

- RCIC received an auto initiation signal on RPV Low-Low Level.

5 minutes later:

- The RCIC pump flow signal input to the flow controller failed HIGH.

Assuming that no operator action has been, or will be taken, complete the following statement describing the RCIC system response.

The RCIC turbine speed will...

A. remain relatively constant.
B. Lower to approximately 2000 rpm.
C. Rise to approximately 4500 rpm and will subsequently trip on high RPV water level.
D. Rise until the mechanical overspeed trips the turbine.

Answer: **B**

DISTRACTOR ANALYSIS

In automatic flow control mode, the RCIC system flow controller compares the pump flow with the controller-setpoint and generates a signal proportional to the difference. Controller output of 4 mA to 20 mA corresponds to turbine speeds of 2000 rpm to 4500 rpm, respectively. In this condition, the RCIC turbine speed would decrease due to the high flow signal and continue to operate at 2000 rpm. The mechanical overspeed would not be reached (5625 RPM). High water level would not be reached because the RCIC turbine is operating at minimum speed.

A. Incorrect, plausible if the applicant believes the controller is normally in manual.
B. **Correct**
C. Incorrect, this would occur if the signal failed low.
D. Incorrect, rpm is limited to 4500 by the governor, which did not fail.

QUESTION No. 39 (page 2 of 2)

APPLICANT COMMENT/CONTENTION

The applicant implies in their comment that answer choice 'C' is also a correct answer.

> *IAW B.02.03-02; a loss of Flow signal will allow RCIC to drive to 4500 RPM, a failed high flow indication will indeed drive RCIC to 2000 RPM; without knowing the size of the leak or even if one exists; RCIC will indeed trip on high level with MSIVs closed and no leak. Given the stated conditions; it could also continue to run at 2000 RPM. Indication or the leak rate could help alleviate confusion. While choice is offered between RPMs, it is entirely probable that the turbine will trip on high level.*

FACILITY RESPONSE AND PROPOSED RESOLUTION

The facility response states that the question is acceptable as written.

> *A leak is not stated to be occurring in the question and would not have any effect on RCIC operation. The correct answer does not address the long term effects of RCIC operation only that RCIC turbine speed will lower to 2000 rpm.*
>
> *Question clarification not requested during exam administration.*
>
> *Question acceptable as written.*
>
> *Reference: B.02.03-02 (RCIC)*

NRC EVALUATION/RESOLUTION

The applicant contends that eventually RCIC will trip on high [RPV water] level and assumes that a leak must be occurring that will eventually lead to RCIC injection. As stated in the Facility Response, the stem does not indicate whether the Group 1 (MSIV) Isolation was due to a steam leak, nor if a leak is continuing after the isolation. Regardless, a Reactor Coolant System leak will not affect how the RCIC flow controller responds to a high failure of the flow input signal. As acknowledged by the applicant, and confirmed by the Facility, a high failure of the flow channel input to the controller will cause the RCIC turbine speed to lower to minimum controller setting (2000 rpm). Speed will not stay the same (answer choice 'A'), nor will it rise (answer choices 'C' and 'D'). Even if RPV conditions were such that RCIC were capable of injecting into the RPV, that does not change the fact that RCIC turbine speed would lower. Only answer choice 'C' states that RCIC turbine speed would lower.

CONCLUSION

Based the information provided and a review of the applicable references, the NRC concludes that the question is acceptable as written, and that the original answer is the correct answer.

QUESTION No. 47

The plant is operating at 100% power when a sudden pressure fault condition occurred in the generator main transformer (Sudden Pressure Relay, SPR-63, actuates)

Which of the following completes the statements below?

This condition DIRECTLY actuates the __(1)__.

The field breaker trips open __(2)__.

A. (1) turbine lockout relay (286/T)
   (2) immediately

B. (1) turbine lockout relay (286/T)
   (2) once 8N7 and 8N8 are sensed open

C. (1) generator lockout relay (286/G)
   (2) immediately

D. (1) generator lockout relay (286/G)
   (2) once 8N7 and 8N8 are sensed open

Answer: **C**

DISTRACTOR ANALYSIS

A. (1) Incorrect, the turbine lockout relay 286/T is actuated by the generator lockout relay286/G.
   (2) Correct, generator lockout relay 286/G trips and locks out the field breaker.

B. (1) Incorrect, the turbine lockout relay 286/T is actuated by the generator lockout relay 286/G when.
   (2) Incorrect, generator lockout relay 286/G trips and locks out the field breaker. Would be correct for any turbine trip not caused by a generator lockout.

C. (1) **Correct**, SPR-63; Generator transformer sudden pressure will operate for fault in the generator transformer and operation of this relay will cause the generator lockout relay 286/G to trip.
   (2) **Correct**, generator lockout relay 286/G trips and locks out the field breaker.

D. (1) Correct, SPR-63; Generator transformer sudden pressure will operate for fault in the generator transformer and operation of this relay will cause the generator lockout relay 286/G to trip.
   (2) Incorrect, generator lockout relay 286/G trips and locks out the field breaker. Would be correct for any turbine trip not caused by a generator lockout.

QUESTION No. 47 (page 2 of 3)

APPLICANT COMMENT/CONTENTION

The applicant contends that there are two correct answers, choices 'C' and 'D'.

> *IAW B.09.02-02; when the 286/G trips; the Turbine Lockout Relay 286/T actuates causing the Field Breaker to open after 8N7 and 8N8. The Field breaker is also listed as a Trip and Lockout when the286/G Trip occurs. There is no mention as to whether this is immediate for SPR-63 actuation rather that [it] occurs and thus is assumed. Because they occur so closely together, it is operationally insignificant. However, as it is written, there is no verbiage in the procedure that specifically cites [the] Field Breaker Trip as immediate. There are potentially two correct answers as the actuation of the SPR does cause 286/T to occur which does cite specifically that the Field Breaker open[s] once 8N7 and 8N8 are sensed open.*

FACILITY RESPONSE AND PROPOSED RESOLUTION

The Facility's position is that there is no correct answer.

From the logic prints, Main Transformer SPR directly causes a 286/G Main Generator Lockout.

- *A 286/G directly causes the Field Breaker to trip*
- *A 286/G directly causes 8N7 and 8N8 to trip open*
- *A 286/G directly causes a 286/T Main Turbine Lockout*
- *A 286/T directly causes the Field Breaker to open if 8N7 & 8N8 are open*

All of the above occur in less than 1 second and could be considered "immediate." They are also directly tied via the logic and could be considered "direct" actions from the SPR.

This would make both options for part 1 of the answer choices correct.

"The field breaker trips open..." once 8N7 and 8N8 are sensed open is a true statement since the question doesn't ask which occurs first. The correct answer for [part 2] should state "directly from a 286/G [actuation]" not just "immediately." This makes both options for part 2 of the answer choices correct.

Question clarification not requested during exam administration.

There is no incorrect answer.

Reference:
B.09.02-02
NE-36013-2
NE-36442-2/3/10

QUESTION No. 47 (page 3 of 3)

NRC EVALUATION/RESOLUTION

The focus of this question is not so much on the time it takes to trip the Main Generator and Main Turbine trips to occur, but the sequence of events and how that sequence differs between a "Generator" trip and a "Turbine" trip.

The 286/G (Generator Lockout) relay protects the Main Generator and Main Transformer from electrical faults (including the Main Transformer Sudden Pressure Relay, SPR-63). The actuation of the 286/G relay immediately (and directly) trips and locks out the generator output breakers (8N7 and 8N8) as well as the generator field breaker to ensure that the electrical are de-energized to isolate the electrical fault. Additionally, the 286/G relay actuation immediately initiates a Main Turbine trip by tripping the 286/T (Turbine Lockout) relay. The Main Turbine is tripped because the complete loss of electrical load will cause the turbine speed to increase leading to a potentially dangerous overspeed condition.

The 286/T (Turbine Lockout) relay primarily protects the Main Turbine from conditions or malfunctions that could damage the Main Turbine. The 286/T relay also provides protection for the Main Generator from conditions that are not an immediate threat to the Main Generator, but that if allowed to continue could lead to generator damage. Actuation of the 286/T relay without actuation of the 286/G relay does not immediately de-energize the Main Generator; the Main Generator remains energized until the Generator Anti-Motor relay is actuated (approximately 2 seconds after the turbine trip) which trips the generator output breakers (8N7 and 8N8), which in turn causes the Generator Field breaker to trip.

Neither the discussion above nor a review of the electrical prints provided support the facility's position that the Turbine Lockout Relay (286/T) is DIRECTLY actuated by the Main Transformer Sudden Pressure Relay (SPR-63). Whether or not the chain of events occurs in less than 1 second is immaterial. Therefore and contrary to the facility's position, only the 286/G option for part 1 of the answer choices (specifically choices 'C' and 'D') is correct.

If the part 2 statement is removed from the context of the question, then the completion of the statement with the phrase, "once 8N7 and 8N8 are sensed open," could be considered a true statement. However, when the statement is completed given the context of the question, and as discussed in the preceding paragraphs and as supported by a review of the provided electrical drawings, the trip of the field breaker, for the scenario given by the question stem (SPR-63 actuation), is NOT dependent upon the position of the generator output breakers (8N7 and 8N8). Therefore, part 2 of answer choices 'B' and 'D' is clearly not true in the context of the question.

The only answer choice that contains the phrases that correctly completes both statements in the question stem, is choice 'C'.

CONCLUSION

Based the information provided and a review of the applicable references, the NRC concludes that the question is acceptable as written, that there is a correct answer, and that the original answer is the only correct answer.

QUESTION No. 48

The plant is at rated conditions when the following occurs:

- 8-A-29 (DIV II INVERTER Y-81 TROUBLE) is received
- The FUSE BLOWN indicator is illuminated on UPS Inverter Panel Y-81
- The Division II 120 VAC UPS System has responded as expected

Which of the following describes how the above conditions will impact components, if at all, associated with the Primary Containment Isolation System (PCIS)?

A. PCIS will be UNAFFECTED.

B. A partial RWCU Group 3 Isolation will occur.

C. The SBGT System will start and Secondary Containment will isolate.

D. The High Temperature Isolation of the RWCU System will be blocked.

Answer: **A**

DISTRACTOR ANALYSIS

A. **Correct:** The inverters will transfer to the alternate source automatically and power to the distribution panels will not be lost. Additionally, placing the MBS in Bypass is Make-Before-Break. Since power is not lost to the distribution panels, components associated with the Containment Isolation System will NOT be affected.

B. Incorrect: The inverters will transfer to the alternate source automatically and power to the distribution panels will not be lost; if power were lost to Y-80, a partial RWCU system isolation would occur.

C. Incorrect: The inverters will transfer to the alternate source automatically and power to the distribution panels will not be lost; if power were to be lost to Y-80, the SBGT System would start and the Secondary Containment would isolate.

D. Incorrect: There is no loss of sync and the inverters will transfer to the alternate source automatically and power to the distribution panels will not be lost; if power were to be lost to Y-30, automatic isolation of the RWCU System on High Filter/Demineralizer Inlet Temperature will be blocked due to loss of power to logic relay.

QUESTION No. 48 (page 2 of 2)

APPLICANT COMMENT/CONTENTION

The applicant contends that there are impacts on PCIS, and that answer choice 'C' is the correct answer.

> *The loss of Y-80 C.4-B.09.13.G specifies the very first Operational Implication as SBGT Starts and Secondary Containment Isolates. The question does not give a specific circuit loss or a total loss, therefore PCIS implications IAW the C.4-B.09.13.G for SBGT and Secondary containment Isolations are applicable unless confirmed to not exist. The omission of C.6-008 -A-14 does not necessarily void SBGT start; is simply states that there is a circuit issue and not a total loss of Y-80. A Group III signal is also plausible to valves MO-2398 and MO-2399 IAW the C.4 for loss of Y-80. There is no definitive conditions in the question that rules out SBGT, Secondary Containment, and RWCU affects as written.*

FACILITY RESPONSE AND PROPOSED RESOLUTION

The facility response states that the question is acceptable as written.

> *Question stem states "The Division II 120VAC UPS System has responded as expected." If the system responds as expected for an Inverter Blown Fuse, Y-30 and Y-80 would not lose power. RWCU, SBGT and Secondary Containment would be unaffected.*
>
> *Question clarification not requested during exam administration.*
>
> *Question acceptable as written.*
>
> *Reference:*
> *8-A-29*
> *B.09.13-02*

NRC EVALUATION/RESOLUTION

The applicant concluded that there was a partial or complete loss of UPS AC Distribution Panel Y-80. As the facility response indicates, IF the Division II 120VAC UPS System responds as expected for an Inverter Blown Fuse, Y-30 and Y-80 would not lose power and PCIS, RWCU, SBGT and Secondary Containment components would be unaffected.

CONCLUSION

Based the information provided and a review of the applicable references, the NRC concludes that the question is acceptable as written, and that the original answer is the correct answer.

QUESTION No. 49

The plant is at rated conditions.

If ALL AC power is lost to LC-107 **and** LC-108; Complete the following statements:

D71 (250 VDC DISTRIBUTION PANEL) is powered from __(1)__.

With no operator action, Y91 (480 VAC UPS) will supply **all** Y94 (480 VAC DISTRIBUTION PANEL) loads for __(2)__ minutes.

A. (1) Battery #17 ONLY
   (2) 30

B. (1) Battery #17 ONLY
   (2) 60

C. (1) Battery #17 AND Y-91
   (2) 30

D. (1) Battery #17 AND Y-91
   (2) 60

Answer: **A**

DISTRACTOR ANALYSIS

A. **(1) Correct**
   (2) Correct

B. (1) Correct
   (2) Incorrect, on a loss of AC power to Y-91, the process computer panel supplies are automatically shed after 30 minutes to extend the availability of 250 VDC Battery 17.

C. (1) Incorrect, the AC source that supplies the battery is from LC-108 to Y-91, LC-107 it the alternate supply to Y-91s output to Y-94.
   (2) Correct

D. (1) Incorrect, the AC source that supplies the battery is from LC-108 to Y-91, LC-107 it the alternate supply to Y-91s output to Y-94.
   (2) Incorrect, on a loss of AC power to Y-91, the process computer panel supplies are automatically shed after 30 minutes to extend the availability of 250 VDC Battery 17.

QUESTION No. 49 (page 2 of 3)

APPLICANT COMMENT/CONTENTION

Although not specifically stated in the applicant's comment, based on the applicant's original answer to the question, the applicant contends that answer choice 'C' is the correct answer.

> *If there is a Loss of Power (AC) to D71, Battery 17 and Y-91 will supply power to the loads. From [Technical Manual] NX-17211 page 43 (attached), "The DC power supplied by either the Rectifier-Charger (Y-91) or the Battery Bank (during emergency operation) is used as input power the Inverter section of the UPS. So the true, technically correct answer is that [the] #17 battery and Y91 power the bus.Y-91 is a Rectifier and Inverter.*

FACILITY RESPONSE AND PROPOSED RESOLUTION

The facility response states that the question is acceptable as written.

> *The first part of the question ONLY asks where D71 is powered from. With no AC power being provided, Battery 17 is the only power source supplying D71.*
>
> *Battery 17 and Y-91 will supply the loads for 30 minutes, but that is NOT what the first part of the question is asking.*
>
> *Clarification was not requested during administration of the exam.*
>
> *Question Acceptable as Written.*
>
> *Reference: B.09.09-02 (UPS)*

NRC EVALUATION/RESOLUTION

Based on the applicant's comment, the applicant has an apparent misunderstanding of what power (AC or DC) is supplied to distribution panel D-71 as well as how USP Y-91 functions. D-71 is a 250 VDC distribution panel and not AC as indicated in the first sentence of the applicant's comments.

Device Y-91 is a 480 VAC uninterruptible power supply (UPS) capable of providing continuous transient free AC power to 480 VAC Distribution Panel Y-94. The major elements of Y-91 are:

a) The Rectifier-Charger section which converts AC power to DC power which is then supplied to the Inverter section of the UPS, as well as to DC Distribution Panel D-71 and Battery 17. The Rectifier-Charger is capable of providing full load support (via the Inverter section) of 480 AC Distribution Panel Y-94, the DC load connected to DCDistribution Panel D-71, and charging of 250 VDC Battery 17.

b) The Inverter section which converts DC power, supplied from either the Rectifier-Charger output (Normal DC Supply) or Battery 17 via distribution panel D-71 (Alternate DC Supply), to 480 VAC power which is then supplied to 480 AC Distribution Panel Y-94 via the Static Switch/Bypass Circuit Breaker.

QUESTION No. 49 (page 3 of 3)

a) Static Switch/Bypass Circuit Breaker section which routes 480 VAC power to 480 AC Distribution Panel Y-94 from either output of the Inverter section (Normal AC Supply) or 480 VAC load center LC-107 (Alternate AC Supply). When a malfunction of the Inverter section is sensed, the Static Switch/Bypass Circuit Breaker section automatically transfers the supply to Y-94 from the Inverter section to the Alternate AC Supply (LC-107).

The DC output from UPS Y-91 to D-71 is only available when 480 VAC power is being supplied to the Rectifier-Charger section from 480 VAC load center LC-108. When LC-108 is de-energized there is no output from the Rectifier-Charger section of UPS Y-91 and the only DC power supply to D-71 will be Battery 17; the answer to part 1 of the question.

As stated in the DISTRACTOR ANALYSIS for the question, and confirmed by the facility response, when all AC input power is lost to UPS Y-91, power to ALL 480 AC Distribution Panel Y-94 loads is continuously maintained for 30 minutes by the UPS Y-91 Inverter output, with the Inverter being supplied from Battery 17 (via D-71). Thirty minutes after power is lost to both LC-107 and LC-108, EDG 13 load shed circuits will trip the Y-94 feeds to the computer distribution panels; the answer to part 2 of the question.

The only answer choice which correctly completes both question statements is choice ‘A’.

CONCLUSION

Based the information provided and a review of the applicable references, the NRC concludes that the question is acceptable as written, and that the original answer is the correct answer.

QUESTION No. 62

A plant startup is continuing following completion of Turbine-Generator roll to 1800 rpm. Prior to Turbine-Generator synchronization, turbine speed lowers and stabilizes at 1550 rpm.

Without further operator action what is an anticipated response, if any?

A. 7-B-33 (TURBINE VIBRATION HIGH) will alarm.

B. Auxiliary Oil Pump will auto start on low oil pressure.

C. No response, this is an expected operating characteristic.

D. 7-B-31/32 (TURB DIFF EXPANSION LONG/SHORT ROTOR) will alarm.

Answer: **B**

DISTRACTOR ANALYSIS

A. Incorrect: this is not considered a vibration-sensitive speed; continuous operation near the turbine's critical speeds of 1150, 1200 and 1400 rpm is not permitted (acceleration should be constant in these regions), as vibrations are expected to be a concern.

B. **Correct**: the turbine is designed for 1800 rpm. If turbine speed is less than 1600 rpm the Main Shaft Oil Pump will be ineffective and cannot supply the proper oil requirements so the Auxiliary Oil Pump will start at this time (AOP is placed in AUTO during startup when shaft speed gets above 1600 rpm).

C. Incorrect: decreasing to 1550 rpm is NOT an expected operational occurrence, the normal operation of the turbine is 1800 rpm.

D. Incorrect: a DECREASE in the turbine speed should NOT affect the general precaution temperature limit (HP turbine first stage bowl temperature rate of change should NOT exceed 150°F/hr) so differential expansion is not a concern; differential expansion is more of a concern during startup.

APPLICANT COMMENT/CONTENTION

The applicant contends that there are two correct answers, answer choices 'A' and 'B'.

IAW the 2167 Plant Startup Procedure, step 114 has the Operator raise the speed of the turbine in preparation for Synchronization (Completing Turbine-Generator Roll). This step precedes step 116 which has the Operator place P-61 in Stop and then auto. It also specifically cites increased turbine Vibrations at 1800 RPM and No Load that could cause Turbine component damage as well as the associates C.6-007-B-33 turbine high vibrations though not expressly stated. If one were at this step vice steps 119 and above, then this is the correct answer. No specific declaration of step in the 2167 is cited. Thus, there are (2) correct answers depending upon where in the 2167 the operator is at.

a) C.6-007-B-33 Turbine Vibrations as a precaution specifically cited above step 114 in the 2167.

b) Auxiliary Oil Pump auto start if past step 116.

QUESTION No. 62 (page 2 of 4)

FACILITY RESPONSE AND PROPOSED RESOLUTION

The facility agrees with the applicant that both answer choices 'A' and 'B' should be accepted as correct answers.

> *Correct answer justification: If turbine speed is less than 1600 rpm the Main Shaft Oil Pump will be ineffective and cannot supply the proper oil requirements so the AOP will auto start on low oil pressure (AOP is placed in AUTO during startup when shaft speed gets above 1600 rpm).*
>
> *The above justification is correct; however, Procedure 2167 (PLANT STARTUP) doesn't specifically state to place the AOP to AUTO from RUN until after the Main Turbine reaches 1800 rpm (Step 116). The turbine would have reached 1800 rpm during performance of Step 114. If the AOP is still in RUN it will not AUTO start.*
>
> *It is not unreasonable for a candidate to assume that Step 116 had NOT been performed yet. In this case, Choice B would not be correct.*
>
> *Additionally, the caution between steps 113 and 114 provides justification that increased vibration could occur. In this case, Choice A would be correct.*
>
> *Question clarification not requested during administration.*
>
> *Accept both choices A and B as correct.*
>
> *Reference: Procedure 2167 (Plant Startup)*

NRC EVALUATION/RESOLUTION

The initial conditions, stated in the first sentence of the question stem, are that "A plant startup is continuing following completion of Turbine-Generator roll to 1800 rpm." The unstated assumption of the question author and reviewers (both Facility and NRC reviewers) was that all the activities associated with rolling the Turbine-Generator to 1800 rpm, including placement of the Turbine Auxiliary Oil Pump (AOP) in a standby configuration, had been completed and that activities associated with Turbine-Generator Synchronization were about to commence.

Both the applicant's and the Facility's response, take the position that returning the Turbine AOP to a standby configuration cannot be performed until after the Turbine-Generator speed reaches 1800 rpm. The "Purpose" section of procedure 2167 contains the following discussion on usage of the procedure.

> *Since no single procedure can address every startup scenario that may be encountered, a degree of flexibility must be provided to address all potential configurations and situations. Although the checklists and procedures associated with startup provide a specific sequence of steps for bringing the plant to full power, it may be prudent to perform certain steps simultaneously.*

QUESTION No. 62 (page 3 of 4)

Reasonable flexibility in the sequence of steps is permissible provided all of the following are satisfied:

- *Steps are NOT omitted.*
- *Steps are performed in the manner described.*
- *Steps that are called for at or prior to reaching specific operating conditions are performed before passing beyond these conditions.*

From a technical standpoint, the Turbine System Description (B.06.01-02) states that whenever the shaft speed is above 1600 rpm, the Shaft Oil Pump is capable of supplying the oil requirements. At that time, the Auxiliary Oil Pump may be shutdown and placed in the AUTO mode to provide backup to the shaft pump, and will start automatically on a sensed decreasing pressure in the operating oil header.

While it is not incorrect to wait until the Turbine-Generator speed reaches 1800 rpm, both the procedural guidance and the system design, support the position that it would be permissible to align the AOP in its standby configuration while the Turbine-Generator was accelerating to 1800 rpm.

Applicants are briefed on the "Policies and Guidelines for Taking NRC Examination" (Appendix E of NUREG 1021) prior to starting the examination. One of the guidelines addresses making assumptions and states in part:

- If you have any questions concerning the intent or the initial conditions of a question, do not hesitate to ask them before answering the question.
- Note that questions asked during the examination are taken into consideration during the grading process and when reviewing applicant appeals.
- When answering a question, do not make assumptions regarding conditions that are not specified in the question unless they occur as a consequence of other conditions that are stated in the question.

As stated above, if the applicant had any question about the initial conditions stated in the question, they simply needed to ask the examination proctor. However, with that being stated, the initial conditions should have been stated more precisely.

Regarding the "caution" statement related to running the Turbine-Generator unloaded, the purpose of the caution is to limit the amount of time that the Turbine-Generator is run unloaded, specifically at the rated speed of 1800 rpm. The flow of steam through the turbine helps to maintain even heating of the internal components. It takes almost no steam flow to maintain the unloaded Turbine-Generator at set speed. Without the benefit of the added steam flow that comes with increasing the load, increased vibration can occur due to rubs and localized heating of Turbine components. While the "caution" specifically address operation at rated speed (1800 rpm), it would be equally applicable at the reduced speed specified in the question. Since the question stem did not mention any time frame associated with operation of the Turbine-Generator at the reduced speed, high vibrations (answer choice 'A') is a likely outcome.

QUESTION No. 62 (page 4 of 4)

CONCLUSION

Based on the above discussion, the NRC concludes that both answer choices ‘A’ and ‘B’ are acceptable answers to the question, and the answer key will be modified accordingly.

QUESTION No. 83

A transient occurred resulting in the following conditions:

- Drywell pressure is 58 psig and slowly rising
- Torus water level is 10 ft. and steady
- Drywell temperature is 320°F and slowly rising
- Drywell radiation is 10 R/hr and slowly rising
- Attempts to spray the drywell have been unsuccessful
- Significant fuel damage is anticipated

You have determined that it is necessary to vent the Primary Containment in accordance with C.5-3505 (VENTING PRIMARY CONTAINMENT).

Given the above information, which one of the following choices identifies:

(1) The recommended vent path?
(2) The desired strategy for venting Primary Containment?

A. (1) SBGT through the 18 inch torus vent (C.5-3505 PART C).
(2) Venting MUST be limited to ONLY the volume required to maintain pressure below the DW pressure limit.

B. (1) SBGT through the 18 inch torus vent (C.5-3505 PART C).
(2) Venting MAY be extended for a period of time to reduce the amount of radioactivity that may have to be released once fuel damage occurs.

C. (1) Hard Pipe Vent (C.5-3505 PART A).
(2) Venting MUST be limited to ONLY the volume required to maintain pressure below the DW pressure limit.

D. (1) Hard Pipe Vent (C.5-3505 PART A).
(2) Venting MAY be extended for a period of time to reduce the amount of radioactivity that may have to be released once fuel damage occurs.

Answer: **D**

DISTRACTOR ANALYSIS

The preferred method for venting Primary Containment is through the Torus and SBGT so that the discharge will be filtered, scrubbed and elevated. However, if SBGT ductwork is in jeopardy of rupturing due to high pressure in containment, >2.9 psig, then the Hard Pipe Vent should be used to minimize potential impacts on Reactor Building equipment from postulated ductwork failure.

Early or extended Primary Containment pressure reduction to limit radioactivity release may be appropriate if: Significant fuel damage is anticipated. Reducing primary containment pressure while the primary containment atmosphere is still relatively clean increases the capacity of the containment to retain fission products. Later releases, after core damage has progressed, may thereby be avoided.

QUESTION No. 83 (page 2 of 4)

A. (1) Incorrect, due to potential to impacts on reactor building equipment
   (2) Incorrect, plausible as in general pressure should only be reduced to maintain below the DW pressure limit, but with significant fuel damage anticipated early or extended releases increase the capacity of containment to retain fission products.

B. (1) Incorrect, see A (1)
   (2) Correct

C. (1) Correct
   (2) Incorrect, see A (2)

D. (1) Correct
(2) Correct

APPLICANT COMMENT/CONTENTION

The applicant contends that answer choice 'C' is also a correct answer, and may be the only correct answer.

Understanding that Significant Fuel Damage is Anticipated; there is a paragraph above the one utilized for this answer in C.5-3505 Bases that states the following: "If the containment atmosphere may be contaminated, the volume released should generally be limited to that required to maintain primary containment pressure below the D'W pressure limit. Release Rate Limits should be exceeded only to the extent necessary to prevent further degradation of plant conditions."

Containment pressurized where it is, is contaminated with 10 R/hr as indicated on Figure 7.1 (Containment Radiation Monitor Response Curves) of the Core Damage Assessment A.2-208.

Without trends as they pertain to the RPV and Containment; we cannot indiscriminately vent the primary per the overarching Bases of C.5-3505 and General Precaution that states the following: "If the containment atmosphere is contaminated, the volume release should be limited to that required to maintain primary containment pressure below the Drywell Pressure Limit (SPDS 81). Release rate limits should be exceeded only to the extent necessary to prevent further degradation of plant conditions."

In addition, the allowance of venting for an extended period of time to reduce the amount of radioactivity from fuel damage actually occurring is followed immediately after with: "may be appropriate while the Primary Containment atmosphere is still relatively clean."

Not knowing the full range of effects from the transient, relatively can only be inferred; thus both answers could be correct. Not having defined values, one can infer that venting via the Hard Pipe vent could challenge the ODCM limits as referenced to those in 10 CFR 50 Appendix I; therefore, limiting venting is more than reasonable.

QUESTION No. 83 (page 3 of 4)

FACILITY RESPONSE AND PROPOSED RESOLUTION

The facility response states that the question is acceptable as written.

Similar conditions were established as follows in the simulator using a Large Steam Line Rupture with loss of Pressure Suppression and 1% Fuel Failure:

- *DW Pressure ~60 psig*
- *DW Temperature ~320°F*
- *DW Radiation ~10 R/hr*
- *DW Contamination ~0.5 uCi/gm*
- *Offsite Dose projection ~1 mr/hr at the site boundary*

Similar conditions were established as follows in the simulator using a Large Steam Line Rupture with loss of Pressure Suppression and 10% Fuel Failure:

- *DW Pressure ~60 psig*
- *DW Temperature ~320°F*
- *DW Radiation ~40 R/hr*
- *DW Contamination ~1.5 uCi/gm*
- *Offsite Dose projection ~10 mr/hr at the site boundary*

Similar conditions were established as follows in the simulator using a Large Steam Line Rupture with loss of Pressure Suppression and 30% Fuel Failure:

- *DW Pressure ~60 psig*
- *DW Temperature ~320°F*
- *DW Radiation ~350 R/hr*
- *DW Contamination ~150 uCi/gm*
- *Offsite Dose projection ~40 mr/hr at the site boundary*

A judgment call would be made by the crew in these circumstances. Based on the information provided above if a small amount of fuel failure is currently occurring (1%) and a more significant amount of fuel failure is expected to occur (30%) the best course of action is to vent longer and earlier to prevent a large late release and thus lower the dose to the public.

Extended venting is allowed if significant fuel damage is anticipated IAW C.5.1-1200, which is stated in the stem of the question.

Question clarification not requested during administration.

Question acceptable as written.

Reference:
C.5.1-1200
C.5-3505
A.2-208

QUESTION No. 83 (page 4 of 4)

NRC EVALUATION/RESOLUTION

Given the initial conditions stated in the question stem, there is no question that the Containment must be vented, only the manner in which the Containment will be vented. The answer to the first part of the question (recommended vent path) is not being challenged, therefore the following discussion will be focused on whether or not the venting duration is required to be limited.

Reducing primary containment pressure increases the capacity of the containment to retain fission products, thereby reducing the amount of radioactivity that may have to be released to the environment. If significant fuel damage is anticipated (as stated in the stem of the question), early or extended venting may be appropriate while the primary containment atmosphere is still relatively clean. The applicant's contention appears to be centered on the contamination level of the Containment atmosphere. The applicant contends that insufficient information is provided to make a judgement on the level of contamination. The primary factor affecting the amount of contamination within the Containment atmosphere is the amount of core damage that has taken place. Since there is no direct way to measure the amount of core damage, it must be determined from indirect methods. The degree of core damage is assessed through the measurement of fission product concentrations in water and gas samples taken from the RCS and Containment, as well as by Containment radiation dose rate measurements and measured Containment Hydrogen concentrations. The easiest and quickest method available to Control Room decision makers to assess the degree of core damage, is by comparing the Containment/Drywell radiation dose rate (given in the stem of the question) to the Containment Radiation Monitor Response Curve (Figure 7.1) in emergency response procedure A.2-208 (Core Damage Assessment). While Figure 7.1 was not provided with the examination, the applicants are expected to have a general understanding of degree of core damage relative to indicated Containment radiation levels. The Drywell radiation dose rate given in the stem of the question (10R/hr) is indicative of a very low level of core damage (also supported by the scenario information provided in the Facility response), and is consistent with a Dose Equivalent Iodine concentration in the reactor coolant that is within or just above the Limiting Condition for Operation (LCO) limit of Technical Specifications.

Given the discussion in the previous paragraph, as well as the initial condition in the stem stating, '"Significant fuel damage is anticipated" (i.e. significant core damage has not yet occurred), it is readily apparent that the Containment atmosphere is "relatively clean.'

With the Containment atmospheric contamination being low and the statement that "Significant fuel damage is anticipated," the applicant's concern, that extended venting could be more detrimental to the public, is unsupportable.

CONCLUSION

Based on the above discussion, the NRC concludes that the original answer choice (D) is the correct, and only correct answer, to the question.

QUESTION No. 97

An EOP Flowchart change to C.5-1200 is being developed IAW 4-AWI-08.16.01 (MONTICELLO EOP AND BEYOND-DESIGN-BASIS GUIDELINE MAINTENANCE PROGRAM) due to a recent change in the BWROG EPGs.

IAW 4-AWI-08.16.01, complete the statements below?

(1) The performance of a 10 CFR 50.59 __(1)__ is required for all EOP changes.

(2) EOP Verification and Validation is required to be performed __(2)__ PORC review.

A. (1) screening
(2) before

B. (1) screening
(2) after

C. (1) evaluation
(2) before

D. (1) evaluation
(2) after

Answer: **A**

DISTRACTOR ANALYSIS

4-AWI-08.16.01 requires a 10 CFR 50.59 screening to be performed for all EOP changes and verification and validation ***SHALL*** be performed prior to submitting the flowchart revision to PORC for review.

A. **Correct**

B. (1) Correct
(2) Incorrect, the verification and validation is required prior to PORC review to ensure technical accuracy and usability.

C. (1) Incorrect, screening is required by procedure.
(2) Correct

D. (1) Incorrect, screening is required by procedure.
(2) Incorrect, the verification and validation is required prior to PORC review to ensure technical accuracy and usability.

QUESTION No. 97 (page 2 of 2)

APPLICANT COMMENT/CONTENTION

The applicant contends that specific knowledge of the contents in the referenced procedure is beyond the level required to be memorized by the applicants.

> *While 4 AWI-08.16.01 does call out that all EOPs will be 50.59 Screened, a screening/evaluation is described in Part III of the QF0022 as it pertains to EOPs in the Site Emergency Plan. The question requires memorization of 4 AWI-08.16.01 whereas knowledge of associated procedures conflicts potential answers given.*

FACILITY RESPONSE AND PROPOSED RESOLUTION

The facility response states that the question is acceptable as written.

> *All EOP changes require a 10 CFR 50.59 screening.*
>
> *EOP changes that affect EAL classifications for the Site Emergency Plan require a 10 CFR 50.54 (q) review.*
>
> *Question acceptable as written.*
>
> *Reference:*
> *4-AWI-08.16.01*
> *QF0022*

NRC EVALUATION/RESOLUTION

Question does not require specific knowledge of the AWI. The applicant is expected to have a working knowledge of the requirements for performing 10 CFR 50.59 screenings and full safety evaluation. This includes having knowledge of types of document changes that must be screened. The applicant is also expected to know the purpose/role of the PORC in the procedure change and approval process.

CONCLUSION

Based the information provided and a review of the applicable references, the NRC concludes that the question is acceptable as written, and that the original answer is the correct answer.

QUESTION No. 100

Which of the following is an INITIAL Protective Action Recommendation (PAR) for a General Emergency Classification made in the Control Room NOT due to a loss of the containment barrier?

A. Evacuate a 2 mile radius and 2-5 mile down wind. All others monitor and prepare.

B. Evacuate a 2 mile radius and 2-10 mile down wind. All others monitor and prepare.

C. Contact the State Planning Chief or State Duty Officer if State EOC is not activated.

D. No protective action recommendation is appropriate when projected plume dose rates do NOT exceed 1000 mrem (TEDE) OR 5000 mrem (CDE) thyroid dose.

Answer: **A**

DISTRACTOR ANALYSIS

A. **Correct:** According to A.2-204, declaration of a General Emergency requires timely initial protective action recommendations (PARs) to off-site agencies. Under these circumstances, NO dose projections are required for formulating the initial off-site protection action recommendation UNLESS there is a Rapidly Progressing Severe Accident. A Rapidly Progressing Severe Accident is a General Emergency (GE) with rapid loss of containment integrity (emergency action levels indicate containment barrier loss) and loss of ability to cool the core. Thus, since this GE is NOT due to loss of containment barrier per the EAL, the applicant shall conclude a GE exists WITHOUT a Rapidly Progressing Severe Accident. Figure 7.3.A of A.2-204 is used. The applicant should conclude the initial PAR is to “Evacuate a 2 mile radius and 2-5 miles downwind. All others monitor and prepare”.

B. Incorrect: 2-10 miles downwind is not the initial PAR distance if a Rapidly Progressing Severe Accident does NOT exist (there is not a loss of containment barrier).

C. Incorrect: This action is taken AFTER the recommendation is made.

D. Incorrect: Initial PARs for a General Emergency involving loss of physical control or core damage require immediate evacuation of the general public and are justified without dose projection; recommending off-site protective actions shall not be delayed.

QUESTION No. 100 (page 2 of 3)

APPLICANT COMMENT/CONTENTION

The applicant contends that no protective action recommendation (PAR) is appropriate (answer choice 'D') since the conditions stated in the question stem do not indicate that any radiological release is occurring or projected to occur.

> *PARs without a loss of containment barrier does not meet the definition for evacuation in procedure A.2-204: "Evacuation is the removal of people from an area to avoid or reduce high-level, short term exposure, from a plume or from deposited activity."*
>
> *A.2-204 Definition second paragraph: "Initial PARs for a General Emergency involving loss of physical control or core damage are based on NRC Response Technical Manual RTM-93, Vol 1, Rev. 3, Section I. Immediate evacuation of the general public is justified without dose projection."*
>
> *Thus, without a loss of a containment barrier to include Primary, Fuel, or RCS and no further information indicative of loss of control, loss of Secondary Containment or that a release is in progress; one cannot justify evacuation, a basis for the General emergency must be declared to evacuate as the evacuation itself could be more dangerous to the public. A more clear indication such as the actual emergency is required to justify evacuation. Recommend adding basis for the General Emergency to satisfy the definition of the evacuation.*

FACILITY RESPONSE AND PROPOSED RESOLUTION

The facility response states that the question is acceptable as written.

> *A General Emergency declaration requires Initial PARs. If a rapidly progressing severe accident does not exist then the PAR would be to evacuate a 2 mile radius and 2-5 miles downwind. All others monitor and repair.*
>
> *Question clarification was requested by candidate as follows:*
>
> > *Question: "What does NOT due to a loss of containment barrier imply?"*
> > *Response: Re-read stem to candidate as written.*
>
> *Question acceptable as written.*
>
> *Reference: A.2-204*

NRC EVALUATION/RESOLUTION

In accordance with the Monticello Nuclear Generating Plant (MNGP) Emergency Plan and the associated implementing procedures, whenever a General Emergency (GE) is declared, regardless of the reason for the GE declaration, Protective Action Recommendations (PARs) shall be made to Local and State authorities.

QUESTION No. 100 (page 3 of 3)

In accordance with Emergency Plan Implementing Procedure A.2-204 (Off-Site Protective Action Recommendations), Control Room decision makers are given two choices (see answer choices 'A' and 'B') to select from. The only difference between the two choices is the outer radius of the downwind sector to be evacuated. The selection is based on the determination of whether or a not the emergency event is a "Rapidly Progressing Severe Accident," which is defined as "a General Emergency (GE) with rapid loss of containment integrity (emergency action levels indicate containment barrier loss) and loss of ability to cool the core." The question stem clearly states that there has not been any loss of the containment barrier, therefore the event is NOT a "Rapidly Progressing Severe Accident," the PAR is as stated in answer choice 'A'.

CONCLUSION

Based the information provided and a review of the applicable references, the NRC concludes that the question is acceptable as written, and that the original answer is the correct answer.

SIMULATION FACILITY FIDELITY REPORT

Facility Licensee: Monticello Nuclear Generating Plant

Facility Docket No: 50-263

Operating Tests Administered: November 14 through 17, 2016

The following documents observations made by the NRC examination team during the initial operator license examination. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance with Title 10 of the *Code of Federal Regulations* 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating tests, the following items were observed:

ITEM	DESCRIPTION
None	

Letter to Peter A. Gardner from Robert J. Orlikowski dated February 7, 2017

SUBJECT: Monticello Nuclear Generating Plant – NRC INITIAL LICENSE EXAMINATION REPORT 05000263/2016301

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